UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

August 27, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: SUMMARY REPORT - 484th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JULY 11-13, 2001 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 484th meeting, July 11-13, 2001, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports and letter. In addition, the Committee authorized Dr. John T. Larkins, Executive Director, ACRS, to transmit the memoranda noted below:

REPORTS

— Recommendation on the Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Report to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS, dated July 20, 2001)

— Circumferential Cracking of PWR Vessel Head Penetrations (Report to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS, dated July 23, 2001)

— South Texas Project Nuclear Operating Company Requests for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations (Option 2) (Report to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS, dated July 23, 2001)

— SECY-01-0100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools" (Report to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS, dated July 20, 2001)

— Feasibility Study on Risk-Informing the Technical Requirements of 10 CFR 50.46 for Emergency Core Cooling Systems (Report to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS, dated July 25, 2001)

LETTER

— Draft NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program" (Letter to William D. Travers, Executive Director for Operations, NRC, from George E. Apostolakis, Chairman, ACRS, dated July 20, 2001)

MEMORANDA

— Draft Regulatory Guide DG-1108, "Combining Modal Responses and Spatial Components in Seismic Response" -- Proposed Revision 2 (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated July 17, 2001)

— Draft Regulatory Guide (DG)-1077, "Guidelines for Environmental Qualification of Microprocessor-Based Equipment Important to Safety in Nuclear Power Plants" (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated July 25, 2001)

— Withdrawal of Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants" (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated July 25, 2001)

HIGHLIGHTS OF KEY ISSUES CONSIDERED BY THE COMMITTEE

1. Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50

The Committee heard presentations by and held discussions with representatives of the NRC staff and the industry concerning the results of the staff's Phase 1 feasibility study for risk-informing 10 CFR 50.46 for emergency core cooling systems (ECCS) and proposed revision to the framework for risk-informing the technical requirements of 10 CFR Part 50. The Committee discussed the staff's request to proceed with rulemaking to modify the existing 10 CFR 50.46 to replace prescriptive ECCS acceptance criteria with a performance-based requirement and to modify the 10 CFR Part 50, Appendix K

evaluation model. The Committee discussed the staff's request to proceed with development of a voluntary risk-informed alternative to 10 CFR 50.46, Appendix K, and General Design Criterion (GDC) 35, "Emergency Core Cooling," of 10 CFR Part 50, Appendix A. The Committee also discussed the staff's proposed longer-term effort to develop the technical bases and requirements for redefining the large-break loss-of-coolant accident (LBLOCA). The Committee considered the staff's proposal to pursue each of these options in parallel.

Committee Action

The Committee issued a report, dated July 25, 2001, to Chairman Meserve on this matter.

2. SECY-01-0100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools"

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning SECY-01-0100. In SECY-01-0100, the staff presented five policy issues and identified a number of options for addressing these issues. These policy issues are related to regulatory decisionmaking in the areas of insurance, emergency preparedness (EP), and safeguards for decommissioning nuclear power plants.

The Nuclear Energy Institute (NEI) representatives briefed the Committee regarding the industry's views on SFP risk study and policy options. NEI recommended that best-estimate evaluations be included in the technical study to promote technology transfer for risk-informed decisions as well as for other studies that could use such information. In addition, NEI recommended a formal peer review of NUREG-1738 to ensure that the relevant experience and experimental insights have been incorporated.

Committee Action

The Committee issued a report to Chairman Meserve, dated July 24, 2001, on this matter.

3. Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"

The Committee heard presentations by and held discussions with representatives of the NRC staff and NEI regarding the need to revise 10 CFR Part 54 to resolve generic

technical issues associated with the license renewal process. The Committee also discussed written comments provided by the Union of Concerned Scientists.

Committee Action

The Committee issued a report to Chairman Meserve, dated July 20, 2001, on this matter.

4. Control Rod Drive Mechanism Cracking

The Committee heard presentations by and held discussions with representatives of the NRC staff and the Electric Power Research Institute Materials Reliability Program regarding industry and staff initiatives to address circumferential cracking of reactor pressure vessel head penetrations, including the control rod drive mechanism nozzles. Based on incidents of cracking discovered during inspections of control rod drive mechanism nozzles at the Oconee and Arkansas Nuclear One, Unit 1 plants, the NRC staff plans to issue a Bulletin with a set of information requests relative to plant-specific susceptibility to cracking, and inspection plans and schedule. The staff expects that the majority of plants most susceptible to cracking will conduct visual inspections during refueling outages scheduled this Fall. Further action by the staff will depend the results of these inspections. Additional work by the staff and the industry is under way relative to crack characterization, inspection methods, and leakage detection.

Committee Action

The Committee issued a report to Chairman Meserve, dated July 23, 2001, on this matter.

5. Draft Individual Plant Examination of External Events Insights Report

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning the draft NUREG-1742, "Perspectives Gained from Individual Plant Examination of External Events (IPEEE) Program." The staff summarized the perspectives gained through the IPEEE Program for external hazards such as seismic events, internal fires, tornadoes, and external floods as well as the Unresolved Safety Issues and Generic Safety Issues (GSIs) that the licensees were requested to address in the IPEEE Program.

These issues are considered resolved on the basis of the information provided by the licensees, with the exception of GSI-172, "Multiple System Responses Program." The

staff plans to develop a resolution package for GSI-172 for ACRS review. The staff stated that the IPEEE Program was generally successful and met the intent of Generic Letter 88-20 Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f).

Committee Action

The Committee issued a letter to the Executive Director for Operations, dated July 20, 2001, on this matter.

6. Status of the Resolution of Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Pump Performance"

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning the status of the resolution of Generic Safety Issue (GSI) 191.

The staff stated that under a technical assistance contract, the Los Alamos National Laboratory performed a study to determine if the transport and accumulation of debris in containment following a loss-of-coolant accident (LOCA) will impede the operation of the emergency core cooling system in operating pressurized water reactors (PWRs). Specifically, the study was to determine whether debris accumulation on sump screens will cause loss of net positive suction head (NPSH) margin following a LOCA and whether further action needs to be taken for PWRs beyond what was done during the resolution of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance."

There were 69 parametric evaluations performed to determine whether sump blockage is a generic concern for PWRs. The results indicated that very little fibrous and particulate debris is needed to cause sump failure. Most of parametric cases analyzed for large LOCA resulted in sump failure.

Committee Action

This briefing was for information only and no Committee action was required on this matter. The Committee plans to review the proposed resolution of GSI-191 during the September 2001 ACRS meeting.

7. Potential Margin Reductions Associated with Power Uprates

The Committee heard a report from the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the meeting of June 12, 2001 during which potential issues associated with power uprates were discussed. In addition, the Committee heard a presentation by and held discussions with representatives of the NRC staff relative to the issue of potential margin reductions associated with core power uprates. The staff stated that for the Duane Arnold plant uprate application, no major concerns have been identified to date. The Office of Nuclear Regulatory Research has instituted a program to investigate potential synergisms arising from core power uprates.

The NRC staff representatives addressed the issue of the need for developing a Standard Review Plan (SRP) Section for power uprate applications. The staff position is that a SRP Section is not needed. Rather, the staff has developed a "template review", based on the experience gained during the review of the Monticello (BWR) and Farley (PWR) power uprate requests.

Committee Action

The Committee decided not to provide formal comment on this issue at this time. This matter will be factored into the Committee's review of plant-specific uprate applications.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated June 11, 2001, to the ACRS comments and recommendations included in the ACRS report dated May 18, 2001, concerning the staff's report on the safety aspects of the license renewal application for Arkansas Nuclear One, Unit 1.

  The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from June 7, 2001, through July 10, 2001, the following Subcommittee meetings were held:

- Thermal-Hydraulic Phenomena - June 12, 2001

The Subcommittee discussed potential issues for consideration by the NRC staff

pertaining to its review of applications for core power uprates.

- Reliability and Probabilistic Risk Assessment - June 22, 2001

The Subcommittee reviewed the staff's draft Individual Plant Examination of External Events Insight Report (draft NUREG-1742).

- Plant Operations and Fire Protection - June 28, 2001

The Subcommittee discussed issues of mutual interest in the areas of fire protection and plant operations.

- Plant Operations - July 9, 2001

The Subcommittee continued its discussion of the Reactor Oversight Process.

- Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment - July 9, 2001

The Subcommittees discussed the proposed risk-informed revisions to 10 CFR 50.46 for emergency core cooling systems and proposed revisions to the framework for risk-informing the technical requirements of 10 CFR Part 50.

- Materials and Metallurgy and Plant Operations - July 10, 2001

The Subcommittees discussed the control rod drive mechanism cracking issues.

- Planning and Procedures - July 10, 2001

The Subcommittee discussed proposed ACRS activities, practices, and procedures for conducting Committee business and organizational and personnel matters relating to ACRS and its staff.

LIST OF MATTERS FOR THE ATTENTION OF THE EXECUTIVE DIRECTOR FOR OPERATIONS

- The Committee plans to discuss the proposed resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance," at the September 2001 meeting.

- The Committee plans to review the technical work and regulatory guidance needed to support the rulemaking effort associated with risk-informing 10 CFR 50.46.

- The Committee would like to receive an update briefing from the staff subsequent to the staff's evaluation of the licensee responses to the bulletin associated with the circumferential cracking of PWR vessel head penetrations.

- The Committee plans to comment on the Unresolved Safety Issues and Generic Safety Issues that were addressed by the IPEEE Program after the staff has responded to the public comments on Draft NUREG-1742, " Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program."

- The staff has committed to provide a resolution package for GSI-172, "Multiple System Responses Program," to the ACRS for review during a future meeting.

- The Committee plans to work with the staff in the development of risk-based performance indicators.

- The Committee plans to review the proposed final revision 1 to Regulatory Guide 1.174, and Standard Review Plan Chapter 19 after reconciliation of public comments.

- The Committee plans to review the proposed final version of Regulatory Guide DG-1077, "Guidelines for Environmental Qualification of Microprocessor-Based Equipment Important to Safety in Nuclear Power Plants," after reconciliation of public comments.

- No EDO response is required to the Committee's July 20, 2001 report regarding the need to revise 10 CFR Part 54.

PROPOSED SCHEDULE FOR THE 485th ACRS MEETING

The Committee agreed to consider the following topics during the September 5-8, 2001, ACRS meeting:

EPRI Report on Resolution of Generic Letter 96-06 Waterhammer Issues
Briefing by and discussions with representatives of the NRC staff and the Electric Power Research Institute (EPRI) regarding the EPRI Report, TR-113594, "Resolution of

Generic Letter 96-06 Waterhammer Issues." [Note: A portion of this session may be closed to discuss EPRI proprietary information.]

Reactor Oversight Process
Briefing by and discussions with representatives of the NRC staff regarding the use of performance indicators in the reactor oversight process, initial implementation of the significance determination process (SDP), and technical adequacy of the SDP to contribute to the reactor oversight process.

Peer Review of PRA Certification Process
Report by an ACRS Senior Staff Engineer regarding the application of the PRA certification process described in NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," for the North Anna Power Station that was conducted by the Westinghouse Owners Group and discussed with the licensee on July 16-20, 2001 in Richmond, Virginia.

Meeting with NRC Commissioner Merrifield
Meeting with Commissioner Merrifield to discuss items of mutual interest.

Proposed Resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance"
Briefing by and discussions with representatives of the NRC staff regarding the proposed resolution of GSI-191.

TRACG Best-Estimate Thermal-Hydraulic Code
Briefing by and discussions with representatives of the NRC staff and the General Electric Nuclear Energy Company regarding the General Electric TRACG best-estimate code and its application to the analyses of anticipated operational occurrences. [NOTE: A portion of this session may be closed to discuss General Electric Proprietary Information.]

The Honorable Richard A. Meserve

Proposed Final Revision to Regulatory Guide 1.78 (DG-1089), "Main Control Room Habitability During a Postulated Hazardous Chemical Release"
Briefing by and discussions with representatives of the NRC staff regarding the proposed final revision to Regulatory Guide 1.78.

Sincerely,

George E. Apostolakis
Chairman

CERTIFIED

Date Issued: 8/27/2001
Date Certified: 9/6/2001

TABLE OF CONTENTS
MINUTES OF THE 484th ACRS MEETING

July 11-13, 2001

ML012530311

REPORTS

— Recommendation on the Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Letter to Chairman Meserve, NRC, from George E. Apostolakis, Chairman, ACRS, dated July 20, 2001)

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APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
V. List of Documents Provided to the Committee

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MINUTES OF THE 484th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS JULY 11-13, 2001 ROCKVILLE, MARYLAND

The 484th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on July 11-13, 2001. Notice of this meeting was published in the *Federal Register* on June 21, 2001 (65 FR 33277) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting was kept and is available in the NRC Public Document Room at the One White Flint North Building, Mail Stop 1F-15, Rockville, MD, 20852-2738. [Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc., 1323 Rhode Island Avenue, NW, Washington, DC 20005-3701, and on the ACRS/ACNW Web page at (www.NRC.gov/ACRS/ACNW).]

ATTENDEES

ACRS Members: ACRS Members: Dr. George Apostolakis (Chairman), Dr. Mario V. Bonaca (Vice Chairman), Dr. F. Peter Ford, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. Dana A. Powers, Mr. Stephen L. Rosen, Dr. William J. Shack, Mr. John D. Sieber, Dr. Robert E. Uhrig, and Dr. Graham B. Wallis. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

Dr. George E. Apostolakis, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee. In addition, it was announced that the Commission appointed Mr. Stephen L. Rosen to the ACRS effective June 13, 2001.

II. Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (Open)

[Note: Mr. Michael T. Markley was the Designated Federal Official for this portion of the meeting.]

Dr. William J. Shack, Chairman of the ACRS Subcommittee on Materials and Metallurgy, introduced this topic to the Committee. He stated that the purpose of this meeting was to discuss the status of proposed risk-informed revisions to 10 CFR 50.46 for emergency core cooling systems (ECCS) and proposed revision to the framework for risk-informing the technical requirements of 10 CFR Part 50.

NRC Staff Presentation

Ms. Mary Drouin, Office of Regulatory Research (RES), led the presentation for the NRC staff. Mr. Alan Kuritsky provided supporting discussion. The staff provided an overview of existing 10 CFR 50.46 requirements and discussed options for developing risk-informed alternative requirements. Mr. Mark Cunningham, RES, provided supporting discussion. Significant points made during the presentation include:

- The staff proposes two short-term options: 1) changes to the current requirements of 10 CFR 50.46 related to acceptance criteria and the evaluation model, and 2) development of a voluntary risk-informed alternative. The staff proposes to revise the current requirements to adopt the more realistic decay heat curve from the 1994 American Nuclear Society (ANS) standard, replace the decay heat multiplier of 1.2 with an NRC-prescribed uncertainty treatment, and replace the Baker-Just zirconium steam model with the Cathcart-Pawel model. The staff also plans to reexamine 10 CFR Part 50, Appendix K to remove unnecessary conservatism. The staff expects to accomplish the technical work for these options in about 12 months depending on Commission approval.

- The staff proposes a longer-term option involving development of the technical bases and requirements for redefining the large-break loss-of-coolant accident (LBLOCA). The staff proposes to continue to evaluate the feasibility of redefining LBLOCA through evaluation of large break frequency, flaw distributions, degradation mechanisms, material response and uncertainty analysis. The staff expects this work may take up to three years.

- Other activities may include revising GDC 35 to replace single failure criterion in the alternative rule, but only as it affects ECCS. The staff plans to defer work on

this until further progress is made on current initiatives to risk-inform 10 CFR 50.46 for ECCS and 10 CFR 50.44 for combustible gas control systems.

- The staff requested an ACRS letter/report on the results of its feasibility study. The staff informed the Committee that the framework document for risk-informing technical requirements of regulations would continue to evolve as more experience is gained, and noted that appropriate consideration of defense in depth would be maintained.

Industry Presentation

Mr. Lewis Ward, Southern Nuclear Operating Company, provided a brief presentation as Chairman of the Westinghouse Owners Group (WOG) LBLOCA reduction project. Significant points made during the presentation include:

- All the Owners Groups support LBLOCA redefinition. Mr. Ward stated that some Owners Groups may have benefits associated with the shorter-term options proposed by the staff. However, LBLOCA redefinition continues to be the item of greatest importance to WOG licensees. Mr. Ward stated that the staff's longer-term schedule for completing the technical bases evaluation for redefinition of LBLOCA is unacceptably slow and WOG may prepare a petition for rulemaking, over the next 6-months, to expedite the process.

- WOG plans to prepare a topical report in support of its rulemaking petition. In that report, WOG plans to examine both pipe size and flow criteria in addressing LBLOCA frequency. Mr. Ward stated that WOG does not propose to eliminate any safety equipment, just relax the requirements on certain items, e.g., safety injection accumulators. He suggested that the technical work for the proposed rulemaking could be accomplished in about a year rather than the three years proposed by the NRC staff.

Dr. Shack questioned what information base is needed to support a redefinition of LBLOCA. The staff stated that there is not much data to support consideration of pipe breaks between 6-inches and a double-ended guillotine pipe break. The staff acknowledged that there may be some graded approaches and that the state of knowledge is not very good. Dr. Shack noted that there is not much service data on large pipe breaks. He also noted that the NRC has previously accepted leak-before-break and probabilistic fracture mechanics methodologies in its assessment of pipe performance. Mr. Rosen stated that data is collected every day through absence of failures.

Dr. Powers questioned the mixing of elements of the Baker-Just and Cathcart-Pawel heat generation and oxidation models. In particular, he expressed concern that the mixing of models may not sufficiently consider model uncertainties and material behavior. He suggested that a performance-based approach be applied so that different fuel cladding materials other than zirconium and ZIRLO could be considered. The staff agreed to consider these suggestions.

Mr. Leitch noted that the industry prefers the long-term option of redefining LBLOCA and questioned what benefits exist for the short-term approach. Staff and industry representatives stated that the Boiling Water Reactor Owners Group plants would benefit substantially from use of the 1994 ANS decay heat curve. The staff stated that this change could be made fairly easily without the use of risk information. Dr. Powers noted that many items proposed by the industry could also be done under risk space without redefining LBLOCA. The staff stated that both short-term options and the longer-term technical work related to redefining LBLOCA are planned as parallel activities to be completed by different organizational units. Thus, progress on the LBLOCA effort is not dependent on the completion of short-term option milestones.

Committee Action

The Committee sent a report on this matter, dated July 25, 2001, to Chairman Meserve.

III. SECY-01-0100, "Policy Issues Related to Safeguards, Insurance and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools"

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning SECY-01-0100. In this SECY, the staff presented five policy issues and identified a number of options for addressing these issues. These policy issues are related to regulatory decision-making in the areas of insurance, emergency preparedness (EP), and safeguards for decommissioning nuclear power plants.

In NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," the staff concluded that a generic decay heat level and decay time beyond which a zirconium fire is physically impossible cannot be defined. The assumption is that the geometry of the spent fuel pool (SFP) assemblies and the associated cooling flow paths are not predictable following a major event that could rupture and rapidly drain the spent fuel pool.

The staff pursued a risk-informed approach for insurance and EP for decommissioning nuclear power plants and initiated the following policy issues:

- Should the Safety Goals for operating nuclear power plants be applied to decommissioning plants?
- Should the Commission develop an approach using probabilistic risk assessments for quantifying the likelihood of sabotage that would permit greater risk-informed regulatory decision making in the area of safeguards?
- How should the Commission define the safeguards protection goal to be applied to spent fuel pools at decommissioning pants?
- What level of insurance is appropriate for licensees of decommissioning plants given the low likelihood of a large onsite and offsite radiological release from a zirconium fire accident involving the spent fuel stored in the spent fuel pool?
- What level of offsite emergency preparedness is appropriate for decommissioning plants given the low likelihood of a radiological release large enough to exceed protective action guides offsite?

Representatives of the Nuclear Energy Institute (NEI) briefed the Committee regarding the industry's views on spent fuel pool risk study and policy options. NEI recommended that best estimate evaluations should be included in the technical study to promote technology transfer for risk-informed decisions as well as for other studies that could use such information. In addition, a peer review is recommended to assure that the relevant experience and experimental insights have been incorporated.

Committee Action

The Committee issued a report to the Commission on this matter dated July 24, 2001.

IV. Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

Mr. Mario Bonaca, Chairman of the Plant License Renewal Subcommittee, stated that the Commission, in a Staff Requirements Memorandum (SRM) dated August 18, 1999,

asked the staff to prepare a detailed analysis and provide recommendations on whether it would be appropriate to resolve generic technical issues by rulemaking.

Mr. Christopher Grimes, Office of Nuclear Reactor Regulation (NRR), introduced the staff presentation. Mr. Sam Lee, NRR, explained that, in response to ACRS comments, the staff had clarified guidance in the license renewal generic guidance documents concerning the inclusion of the results of the scoping process in license renewal applications. He also discussed the staff's disposition of comments presented in a Union of Concerned Scientists' (UCS) letter dated June 26, 2001. Mr. Lee concluded that rulemaking was not necessary and noted that the staff would continue to monitor license renewal lessons learned and other rulemaking activities for opportunities to improve the license renewal process.

Mr. Alan Nelson, NEI, stated that after the approval of the extension of three licenses, the license renewal process was stable and predictable. He concluded that there was no need for rulemaking.

The ACRS Members and the staff discussed the UCS's comments, the NRC petition for rulemaking process, lessons learned from the review of the first boiling water reactor license renewal application, and changes to the Generic Aging Lessons Learned (GALL) report associated with the inspection of small bore piping.

Committee Action

The Committee issued a letter to Chairman Meserve on this matter dated July 20, 2001.

V. Control Rod Drive Mechanism (CRDM) Cracking

[Note: Mrs. Maggalean W. Weston was the Designated Federal Official for this portion of the meeting.]

Dr. Ford, cognizant ACRS Member for this issue, introduced this topic to the Committee. He said that the Committee would be briefed by NRR and Industry Representatives regarding cracking incidents seen to date as well as the NRR's proposal to issue a bulletin requesting information from potentially affected plant licensees. Dr. Ford said that the staff has requested formal Committee comment on this matter at this time.

NRC Staff Presentation

Prior to beginning his presentation, Mr. J. Strosnider, NRR, noted that the staff had taken the action of issuing a generic letter in 1997, subsequent to the discovery of pressure vessel head penetration nozzles in French PWRs. In response, Industry began routine inspections to monitor for leakage. Initially, observed cracking was axially-oriented; some recently-discovered cracks were circumferential, which are of greater risk significance.

NRR addressed the following issues pertaining to CRDM nozzle cracking:

- Safety Perspective
- Technical Issues Highlighted by the ACRS Subcommittees
- Industry and NRC Bulletin Approaches to Inspection
- Risk Assessment
- Additional Work Required
- Relationship of Issue to Agency Performance Goals

Key points noted by Mr. Strosnider included:

- Failure of a CRDM nozzle is not expected to challenge containment integrity.

- Timely, effective inspections should provide confidence that safety is and regulatory requirements are maintained.

- The Bulletin, in requesting information from the industry, suggests a graded inspection approach with four "bins" of affected plants, based on the (currently understood) degree of susceptibility to cracking. The information requested will support assessment of the need for additional regulatory actions. The industry approach to this issue differs in such areas as the scope and timing of inspections and the number of plants affected (25 vs 45 per the NRC).

- Additional work remaining includes: (1) completion of RES' Expert Group's activities, (2) RES response to a NRR user need request for assistance regrading NDE/ISI and issues associated with crack growth, residual stresses, leak detection, and repair and mitigation, and (3) work associated with risk insights and sequence delineation.

In response to a question from Dr. Powers, Mr. Strosnider indicated that preliminary calculations of the conditional core damage probability for a CRDM failure event range from 10^{-3} to 10^{-2}, lead the staff to conclude that additional attention to this matter (i.e., issuance of a bulletin) was necessary.

Electric Power Research Institute Presentation (EPRI)

Mr. L. Matthews, EPRI Materials Reliability Program, presented the industry positions on this matter. Points noted by Mr. Matthews included:

- There is reasonable assurance that PWRs do not have circumferential cracking that would exceed structural margin.
- EPRI has established an activity schedule that calls for its Expert Panel on Crack Growth to meet in early-August, inspections of affected plants conducted during the Fall refueling outages, and development of a reactor pressure vessel safety assessment in December 2001.
- Other on-going activities include work on probabilistic risk assessments, probabilistic fracture mechanics, nondestructive examination demonstration, development of a training package for visual examination, and review of repair and mitigation strategies.

In response to questions from Dr. Powers, Mr. Matthews noted that there was a considerable margin with respect to time (several years) before a CRDM ejection event was threatened. Mr. Matthews also stated that the probability of a multiple ejection event is remote, given the ruggedness of the material involved. Finally, it was noted that the Oconee licensee plans to replace the vessel heads of its three units to address this issue.

Committee Action

The Committee issued a report to Chairman Meserve on this matter dated July 23, 2001.

VI. Draft Individual Plant Examination of External Events (IPEEE) Insights Report

[Note: Mr. Amarjit Singh was the Designated Federal Official for this portion of the meeting.]

Dr. George E. Apostolakis, Chairman of the Subcommittee on Reliability and Probabilistic Risk Assessment, introduced the topic to the Committee. He stated that the purpose of this session was to discuss the staff's draft NUREG-1742, Vols 1 and 2, "Perspectives Gained From Individual Plant Examination of External Events (IPEEE) Program."

NRC Staff Presentation

Mr. Allan Rubin led the discussion for the staff. He summarized the perspectives gained through the IPEEE program and IPEEE related Unresolved Safety Issues (USIs) and Generic Safety Issues (GSIs). The licensees were specifically requested to address the following Issues:

- USI A-45, "Shutdown Decay Heat Removal Requirements"
- GSI-103, "Design for Probable Maximum Precipitation"
- GSI-131, "Potential Seismic Interaction Involving the Movable Incore Flux Mapping Systems Used in Westinghouse Plants"
- GSI-57, "Effects of Fire Protection Systems Actuation on Safety-Related Equipment"
- Sandia Fire Risk Scoping Study issues

These issues are considered resolved on the basis of the information provided by the licensees. GSI-172, "Multiple System Response Program is till open. IPEEE submittals verified 80% of the plants have adequately addressed the IPEEE aspects of this issue. A resolution package will be generated for this issue for ACRS review. It was stated that overall IPEEE was successful and met the intent of Generic Letter 88-20, Supplement 4.

Committee Action

The Committee wrote a letter to the Executive Director for Operations on this matter dated July 20, 2001.

VII. Proposed Resolution of Generic Safety Issues (GSI)-191 "Assessment of Debris Accumulation on PWR Sump Pump Performance"

[Note: Mr. Amarjit Singh was the Designated Federal Official for this portion of the meeting.]

Mr. Graham M. Leitch, Chairman of the ACRS Subcommittee on Plant Systems introduced the topic to the Committee. He stated that the purpose of this session was to discuss the status of the proposed resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance."

NRC Staff Presentation

Mr. Michael Marshall led the discussion for the staff. He presented a brief background and the proposed resolution and status of GSI 191. He stated that under the technical assistance contract the Los Alamos National Laboratory performed the GSI-191 study to determine if the transport and accumulation of debris in containment following a loss-of-coolant accident (LOCA) will impede the operation of the emergency core cooling system (ECCS) in operating pressurized water reactors (PWRs). The purpose of the study was to determine whether debris accumulation on sump screens will cause loss of net positive suction head (NPSH) margin following a LOCA. Also, to determine if further action needs to be taken for PWRs beyond what was done during the resolution of Unresolved Safety Issue (USI) A-43.

Sixty-nine parametric cases were developed for this evaluation provided there was a reasonable representation of operating PWRs, so the results form a credible technical basis for making a determination of whether sump blockage is generic concern for PWRs. However, the parametric evaluations suffer from a number of limitations that make them ill suited for making a determination of whether a specific plant is vulnerable to sump failure. The results indicated that very little fibrous and particulate debris is needed to cause sump failure. Most of the parametric cases analyzed for large LOCA resulted in sump failure.

Committee Action

This briefing was information only and no Committee action is required on this matter.

VIII. Potential Margin Reductions Associated with Power Uprates

[Note: Mr. Paul Boehnert was the Designated Federal Official for this portion of the meeting.]

Dr. Wallis, Chairman of the Thermal-Hydraulic Phenomena Subcommittee, provided the Committee with a report regarding the meeting of June 12, 2001, and said that a presentation from representatives of the NRC staff would be provided relative to the issue of potential margin reductions associated with core power uprates. Regarding the subcommittee meeting, he noted that:

- The staff indicated that for the Duane Arnold plant uprate application, no major concerns have been identified to date. Plant risk is increased slightly, mostly

due to decreased operator response time for ATWS events. Staff audits of the General Electric codes has identified one issue that needs to be addressed, i.e., the use by GE of a code to generate a data base in support of the GEXL-14 critical heat flux correlation, which appears to be a questionable procedure.

- RES has instituted a program to investigate potential synergisms arising from core power uprates. This program appears to be rather extensive in scope, but may not provide results that are timely to the already on-going uprate reviews.

- ACRS Senior Fellow A. Cronenberg reiterated the results of his investigation of the potential for margin reductions associated with core power uprates. Dr. Cronenberg recommends that the NRC develop a Standard Review Plan (SRP) Section for uprate reviews, Legacy Tables should be developed to track plant conditions impacted by such cumulative licensing actions as power uprates, license renewal, etc., and risk assessments should be conducted for significant (15-20%) power uprates.

NRC Staff Presentation

The NRC staff representatives addressed the issue of the need for development of a Standard Review Plan Section for power uprate applications. Issues discussed included Background, Current Guidance, Potential Changes to Review Processes, and Conclusions. Key points noted by the staff included the following:

- Following the concerns identified by the Maine Yankee Lessons Learned Task Group, NRR committed, in 1997, to develop a Standard Review Procedure for power uprate applications.

- The staff now believes that the use of approved GE generic topical reports, applicable SRP Sections, and the Safety Evaluations for the Monticello (BWR) and Farley (PWR) plants acting as review "templates," has rendered unnecessary development of a SRP Section on power uprates.

- Review of the Duane Arnold, Dresden, and Quad Cities extended uprates are seen as first-of-a-kind and the staff intends to conduct a Lessons Learned Workshop, evaluate the review processes to gain efficiencies, and issue written guidance to licensees contemplating similar uprate submittals.

- The uprate review process is still changing, as noted above, and the staff will reevaluate the need for development of a SRP Section in the future.

In response to Dr. Ford, NRR noted that the staff's review on the recently submitted GE Topical Report on Constant Pressure Power Uprate has been discontinued; GE will revise this report. Dr. Bonaca asked if the staff is evaluating plant equipment with regard to the impact on operational duty resulting from synergistic effects (e.g., power uprate and license extension). NRR said that they do consider this issue, both in the review itself and via the inspection and oversight program. In response to Mr. Leitch, NRR said that the staff has addressed the Maine Yankee concerns regarding the necessary scope and depth of review for power uprates.

Committee Action

The Committee decided not to provide formal comment at this time. This matter will be factored into the Committee's review of plant-specific uprate applications.

IX. Executive Session (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

A. Reconciliation of ACRS Comments and Recommendations

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated June 11, 2001, to the ACRS comments and recommendations included in the ACRS report dated May 18, 2001, concerning the staff's report on the safety aspects of the license renewal application for Arkansas Nuclear One, Unit 1.

The Committee decided that it was satisfied with the EDO's response.

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

B. Report on the Meeting of the Planning and Procedures Subcommittee (Open)

- Review of the Member Assignments and Priorities for ACRS Reports and Letters for the July ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the July ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were also discussed.

- Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through October 2001 was discussed. The objectives were:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

- Quadripartite Meeting Update

During the April meeting, the Committee was informed that Mr. Lothar Hahn, Chairman of the RSK, was preparing for the next Quadripartite Meeting that would be hosted by Germany. The French GPR have confirmed their participation and the RSK is currently working to confirm the participation of the Japanese NSC.

During the June 2001 meeting, the Committee proposed the following topics for the Quadripartite meeting:

- Risk-Informed Regulation
- Thermal-Hydraulic Analysis and Code Issues
- High Burnup Fuel
- Risk Analysis of Spent Fuel Storage

The Committee suggested that other countries (e.g., Sweden and Switzerland) be invited to attend this meeting. RSK has informed Dr. Larkins that they plan to discuss the ACRS suggestion with other Quadripartite member countries. Also, the RSK proposed having the Quadripartite meeting during the first full week in June 2002. Because of the anticipated conflict with the ACRS meeting, it was suggested that the Quadripartite meeting be held on June 24-28, 2002 in Berlin, Germany.

- Tour of the Shipyard in Groton, CT, and a Submarine

The ACRS plans to review the new nuclear propulsion plant submarine design (VIRGINIA Class, successor to the LOS ANGELES Class) in 2002. In connection with

this review, the members visited the Naval Reactor (NR) Organization Headquarters Office in Crystal City, Virginia, on April 4, 2000. On August 7, 2000, the members visited the NR training complex located at the Charleston, SC, Naval Base. Recently, representatives of NR discussed with Dr. Apostolakis about potential options for the Committee's review of the VIRGINIA Class submarine as well as a tour of the shipyard construction site in Groton, CT, in November 2001 and tour a submarine in early 2002.

- Revised Subcommittee Structure

A revised ACRS Subcommittee structure was approved by Dr. Apostolakis, ACRS Chairman, and was sent to all members on June 18, 2001.

- Member Requests for Support Services

Whenever members request the ACRS Office to set up an arrangement for support services (e.g., postage, storage space, rental of office space) at their off-site location and subsequently decide that they do not need the requested service(s), the member(s) should inform the appropriate Operations Support staff person as soon as possible of their decision so that we may cancel the arrangement. Otherwise, money that could be more appropriately used to fund office travel demands or other needs remains committed to provide that service and is lost beyond retrieval for the office use after September 30 each year.

- Availability of Business Cards

The EDO has recently authorized the purchase of business cards for employees who perform representational duties requiring them to interact with or conduct NRC business and/or meetings with outside entities. The cards are printed in blue or black ink and there are two layout styles from which to choose. Card quantities of 250 or more must be ordered. We will print smaller orders in house on perforated business card stock.

- Inadvertent Release of Documents to the Public

On June 19, 2001, the NRC discovered that approximately 800 documents stored in the ADAMS (Agency Wide Documents Access and Management System) Main Library marked as "non-public" were inadvertently made available to the public. Some of the documents were site access authorization letters from NRC to various licensees which contain privacy act information for certain NRC employees, including the members who have participated in site visits since 1999.

Members whose information was released have received a letter from the Executive Director of Operations, explaining how the problem came about and what remedial measures have been taken to preclude recurrence of this situation.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 485th ACRS Meeting, September 5-8, 2001.

The 484th ACRS meeting was adjourned at 6:00 p.m. on Friday, July 13, 2001.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 6, 2001

MEMORANDUM TO: Sherry Meador, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFIED MINUTES OF THE 484th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JULY 11-13, 2001

I certify that based on my review of the minutes from the 484th ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

George E. Apostolakis, Chairman

September 6, 2001
Date

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

August 27, 2001

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary Sherry Meador

SUBJECT: PROPOSED MINUTES OF THE 484th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - JULY 11-13, 2001

Enclosed are the proposed minutes of the 484th meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate.

Attachment:
As stated

ague (Blue Ridge); and (4) Charles and dna Foster. The requests were filed in response to a March 7, 2001 notice in the **Federal Register** announcing receipt of the CAR and a related environmental report (66 FR 13794.) and a notice of opportunity for a hearing published by the Commission on April 18, 2001 (66 FR 19994).

The Board is comprised of the following administrative judges:

Administrative Judge Thomas S. Moore, Chairman, Atomic Safety and Licensing Board Panel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555–0001

Administrative Judge Charles N. Kelber, Atomic Safety and Licensing Board Panel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001

Administrative Judge Peter S. Lam, Atomic Safety and Licensing Board Panel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555–0001

All correspondence, documents, and other materials shall be filed with the Administrative Judges in accordance with 10 CFR 2.1203.[1]

Issued at Rockville, Maryland, this 15th day of June 2001.

. Paul Bollwerk III,

Chief Administrative Judge, Atomic Safety and Licensing Board Panel.

[FR Doc. 01–15633 Filed 6–20–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards will hold a meeting on July 11–13, 2001, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Friday, November 17, 2000 (65 FR 69578).

Wednesday, July 11, 2001

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10:00 a.m.: Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed risk-informed revisions to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and proposed revisions to the framework for risk-informing the technical requirements of 10 CFR Part 50.

10:20 a.m.–12:00 Noon: SECY–01–0100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools" (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding SECY–01–0100 and related matters.

[Note: A portion of this session may be closed to discuss safeguards information.]

1:00 p.m.–2:00 p.m.: Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding the need to revise 10 CFR Part 54.

2:15 p.m.–3:45 p.m.: Control Rod Drive Mechanism (CRDM) Cracking (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and NEI regarding the staff and industry proposals for dealing with CRDM cracking.

[Note: A portion of this session may be closed to discuss proprietary information.]

3:45 p.m.–4:15 p.m.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports for consideration by the full Committee.

4:15 p.m.–7:00 p.m.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting as well as a proposed report on South Texas Project Exemption Request.

Thursday, July 12, 2001

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–9:45 a.m.: Draft Individual Plant Examination of External Events (IPEEE) Insights Report (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft IPEEE Insights Report (NUREG–1742).

10:00 a.m.–11:00 a.m.: Proposed Resolution of Generic Safety Issues (GSI)–191, "Assessment of Debris Accumulation on PWR Sump Pump Performance" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the status of resolution of GSI–191.

11:15 a.m.–12:15 p.m.: Potential Margin Reductions Associated with Power Uprates (Open)—The Committee will hold discussions with representatives of the NRC staff regarding ongoing or proposed staff activities related to the development of a Standard Review Plan for use in the review of power uprate applications.

1:15 p.m.–2:00 p.m.: Reactor Oversight Process (Open)—The Committee will discuss a proposed response to the following items in the April 5, 2000 Staff Requirements Memorandum:

- Review the use of performance indicators (PIs) in the Revised Reactor Oversight Process (RROP) to ensure that the PIs provide meaningful insights into aspects of plant operation that are important to safety.
- Review the initial implementation of the significance determination processes (SDPs) and assess the technical adequacy of the SDP to contribute to the RROP.

2:00 p.m.–2:45 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

2:45 p.m.–3:00 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

3:45 p.m.–7:00 p.m.: Discussion of Proposed ACRS Reports (Open)—The

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[1] Copies of this Licensing Board establishment notice were sent this date by internet e-mail or facsimile transmission to counsel or representatives for (1) applicants Duke, Cogema, and Stone & Webster; (2) petitioners GANE, EI, Blue Ridge, and Charles and Edna Foster; and (3) the NRC staff.

Committee will discuss proposed ACRS reports.

Friday, July 13, 2001

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)

8:35 a.m.–5:30 p.m.: Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

5:30 p.m.–6:00 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 11, 2000 (65 FR 60476). In accordance with these procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting and questions may be asked only by members of the Committee, its consultants, and staff.Persons desiring to make oral statements should notify Mr. Howard J. Larson, ACRS, five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting Mr. Howard J. Larson prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with Mr. Howard J. Larson if such rescheduling would result in major inconvenience.

In accordance with Subsection 10(d) P.L. 92–463, I have determined that it is necessary to close portions of this meeting noted above to discuss safeguards information per 5 U.S.C. 552b(c)(3) and proprietary information per 5 U.S.C. 552b(c)(4).

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting Mr. Howard J. Larson (telephone 301–415–6805), between 7:30 a.m. and 4:15 p.m., EDT.

ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the internet at http://www.nrc.gov/ACRSACNW.

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., EDT, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: June 15, 2001.

Annette Vietti-Cook,

Acting Advisory Committee Management Officer.

[FR Doc. 01–15632 Filed 6–20–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR WASTE TECHNICAL REVIEW BOARD

Notice of Meeting; Yucca Mountain, NV, Repository

Board Workshop: July 19–20, 2001—Arlington, Virginia: The Board-sponsored workshop will provide a forum for invited international participants to discuss key issues related to the corrosion of materials being proposed by the Department of Energy for use in waste packages in a potential Yucca Mountain Repository.

Pursuant to its authority under section 5051 of Public Law 100–203, Nuclear Waste Policy Amendments Act of 1987, on Thursday, July 19, and Friday morning, July 20, 2001, the Nuclear Waste Technical Review board (Board) will host a workshop in Arlington, Virginia. The workshop will focus on issues related to long-term (thousands of years) extrapolation of the corrosion resistance of waste package materials being proposed by the U.S. Department of Energy (DOE) for use in a potential Yucca Mountain repository. Approximately 15 corrosion experts from around the world have been invited to participate in the workshop. The Board has long emphasized the importance of issues related to predicting long-term waste package performance. Progress in understanding fundamental corrosion processes has been identified by the Board as one of four priority areas that would be important for a potential Yucca Mountain site recommendation.

The DOE is characterizing a site at Yucca Mountain, Nevada, as the possible location of a permanent repository for spent nuclear fuel and high-level radioactive waste. The Board is charged by Congress with reviewing the technical and scientific validity of DOE activities related to managing spent nuclear fuel and high-level radioactive waste.

The workshop will be held at the Hilton Arlington & Towers; 950 North Stafford Street; Arlington, Virginia 22203. The telephone number is (703) 528–6000; the fax number is (703) 812–5127. The workshop, which is open to the public, will start at 8:30 a.m. on both days and end at approximately 5:30 p.m. on Thursday and 12:00 noon of Friday.

Workshop participants will be given two questions to consider. The questions will seek opinions on possible modes of waste package corrosion that may not be relevant in the usual time frame of engineering experience, but that could develop or accelerate after long periods in a repository. (The questions will be posted on the Board's Web site by July 1.)

The session on Thursday, July 19, will begin with a brief overview of the materials that the DOE proposes for the waste packages and of the thermal and chemical environments that the packages would likely experience over millennia in a possible Yucca Mountain repository. Following the overview, the workshop participants will give short presentations on their initial responses to the two questions. The balance of the workshop will consist of informal roundtable discussions focusing on the two questions. The workshop will be a "brainstorming" session designed to furnish the Board with a diversity of highly qualified opinions on the issue of extrapolating corrosion resistance over very long periods, a key factor in repository performance. A by-product of the workshop may be information that will complement a peer review that the DOE recently initiated to examine a much broader range of corrosion issues.

Opportunities for public comment will be provided before the end of Thursday's session and before adjournment of Friday. Those wanting to speak during the public comment periods are encouraged to sign the "Public Comment Register" at the check-in table. A time limit may have to be set on individual remarks, but written comments of any length may be submitted for the record.

APPENDIX II

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 14, 2001

SCHEDULE AND OUTLINE FOR DISCUSSION
484th ACRS MEETING
July 11-13, 2001

WEDNESDAY, JULY 11, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open)
- 1.1) Opening statement (GEA/JTL/HJL/SD)
- 1.2) Items of current interest (GEA/SD)
- 1.3) Priorities for preparation of ACRS reports (GEA/JTL/SD)

2) 8:35 - ~~10:00~~ 10:20 A.M. Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (Open) (WJS/GBW/MTM)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff regarding proposed risk-informed revisions to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and proposed revisions to the framework for risk-informing the technical requirements of 10 CFR Part 50.

Representatives of the nuclear industry will provide their views, as appropriate.

~~10:00~~ 10:20 - ~~10:20~~ 10:40 A.M. ***BREAK***

3) ~~10:20~~ 10:40 - 12:00 Noon SECY-01-0100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools" (Open/Closed) (TSK/MME)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff regarding SECY-01-0100 and related matters.

Representatives of the nuclear industry will provide their views, as appropriate.

[NOTE: A portion of this session may be closed to discuss safeguards information.]

~~12:00~~ 12:45 - ~~1:00~~ 1:30 P.M. ***LUNCH***

4) 1:30 - 1:50 ~~1:00~~ - ~~2:00~~ P.M. Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Open) (MVB/SD)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussion with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding the need to revise 10 CFR Part 54.

1:50 ~~2:00~~ - **2:15 P.M. ***BREAK*****

5) 2:15 - 4:05 ~~3:45~~ P.M. Control Rod Drive Mechanism (CRDM) Cracking (Open/Closed) (FPF/JDS/MWW/PAB)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and NEI regarding the staff and industry proposals for dealing with CRDM cracking.

~~[NOTE: A portion of this session may be closed to discuss proprietary information.]~~

6) 4:05 - 4:35 ~~3:45~~ - ~~4:15~~ P.M. Break and Preparation of Draft ACRS Reports (Open)
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

7) 4:35 - 7:30 ~~4:15~~ - ~~7:00~~ P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

7.1) Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (WJS/GBW/MTM)

7.2) SECY-01-0100 and Related Matters (TSK/MME)

7.3) South Texas Project Exemption Request (JDS/GEA/MWW)

7.4) Need to Revise 10 CFR Part 54 (MVB/SD)

7.5) Proposals for Dealing with CRDM Cracking (FPF/JDS/MWW/PAB)

THURSDAY, JULY 12, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

8) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GEA/JTL/SD)

9) 8:35 - 9:50 ~~9:45~~ A.M. Draft Individual Plant Examination of External Events (IPEEE) Insights Report (Open) (GEA/AS)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding the draft IPEEE Insights Report (NUREG-1742).

Representatives of the nuclear industry will provide their views, as appropriate.

9:50-10:05
~~9:45~~ - ~~10:00~~ **A.M.** *****BREAK*****

10:05-11:05
10) ~~10:00~~ - ~~11:00~~ A.M. Proposed Resolution of Generic Safety Issues (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance" (Open) (GML/AS)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding the status of resolution of GSI-191.

11:05
~~11:00~~ - **11:15 A.M.** *****BREAK*****

12:23
11) 11:15 - ~~12:15~~ P.M. Potential Margin Reductions Associated with Power Uprates (Open) (GBW/MVB/AWC/PAB)

11.1) Remarks by the Subcommittee Chairman

11.2) Discussions with representatives of the NRC staff regarding ongoing or proposed staff activities related to the development of a Standard Review Plan for use in the review of power uprate applications.

12:23-1:30
~~12:15~~ - ~~1:15~~ **P.M.** *****LUNCH*****

1:30 - 1:50
12) ~~1:15~~ - ~~2:00~~ P.M. Reactor Oversight Process (Open) (JDS/GEA/MWW)

12.1) Remarks by the Subcommittee Chairman

12.2) Discussion of proposed response to the following items in the April 5, 2000 Staff Requirements Memorandum:

- Review the use of performance indicators (PIs) in the Revised Reactor Oversight Process (RROP) to ensure that the PIs provide meaningful insight into aspects of plant operation that are important to safety.
- Review the initial implementation of the significance determination processes (SDPs) and assess the technical adequacy of the SDP to contribute to the RROP.

1:50-2:30
13) ~~2:00~~ - ~~2:45~~ P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

13.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

13.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

2:30-2:35
14) ~~2:45~~ - ~~3:00~~ P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)

Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

3:00 - 3:45 P.M. *****BREAK*****

15) 3:45 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:
15.1) Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50 (WJS/GBW/MTM)
15.2) SECY-01-0100 and Related Matters (TSK/MME)
15.3) South Texas Project Exemption Request (JDS/GEA/MWW)
15.4) Need to Revise 10 CFR Part 54 (MVB/SD)
15.5) Draft IPEEE Insights Report (GEA/AS)
15.6) Proposals for Dealing with the CRDM Cracking (FPF/JDS/MWW/PAB)
~~15.7) Potential Margin Reductions Associated with Power Uprates (tentative) (GBW/MVB/PAB/AWC)~~

FRIDAY, JULY 13, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

16) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GEA/JTL/SD)

17) 8:35 - 5:30 P.M. Proposed ACRS Reports (Open)
(12:00-1:00 P.M. LUNCH) Continue discussion of proposed ACRS reports listed under Item 15.

18) 5:30 - 6:00 P.M. Miscellaneous (Open) (GEA/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- **Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.**
- **Number of copies of the presentation materials to be provided to the ACRS - 35.**

APPENDIX III: MEETING ATTENDEES

484TH ACRS MEETING
JULY 11-13, 2001

NRC STAFF (July 11, 2001)

A Hiser, NRR
K. Wichman, NRR
M. Drouin, RES
A. Kuritsky, RES
W. Beckner, NRR
I. Schoenfeld, OEDO
S. Lee, NRR
A. Levin, OCM/RAM
B. Huffman, NRR
R. Sullivan, NRR
G. Hubbard, NRR
P. Ray, NRR
T. Collins, NRR
R. Temps, NMSS
P. Brachman, NMSS
S. West, NRR
B. Mendelsohn, NMSS
M. Weber, NMSS
L. Pittiglo, NMSS
M. Blevins, NMSS
V. Ordaz, NRR
B. Zalcman, NRR
W. Liu, NRR
R. Elliott, NRR
L. Abramson, RES
J. Zimmerman, NRR
J. Beall, OCM/EM
S. Dinsmore, NRR
J. Lara, OCM/RAM
J. Medoff, NRR
J. Chung, NRR
B. Caldwell, NRR
R. Franovich, NRR
A. Buslik, RES
M. Reinhart, NRR
D. Jackson, RES
W. Norris, RES
J. Tappert, NRR
F. Cherny, RES
B. Bateman, NRR
I. Dinitz, NRR
B. Manili, NRR
B. Skelton, NRR
B. Schnetzler, NRR
B. Palla, NRR
F. Gillespie, NRR
G. Bagchi, NRR
K. Gibson, NRR
J. Uhle, RES
G. Galleti, NRR
J. Vora, RES
K. Rico, NRR
B. Thomas, NRR
J. Carrasco, NRR
S. Lee, NRR
P. T. Kuo, NRR
C. Grimes, NRR
P. Kang, NRR
J. Stenisk, NRR
S. Hoffman, NRR
S. Mitra, NRR
S. Koenick, NRR
B. McCabe, OCM/JSM
R. Hernandez, NRR
D. O'Neal, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

L. Ward, WOG/Southern Nuclear
R. Graybed, Protopower
P. Negus, GE
R. Janati, PA DEP/BRP
R. Huston, Licensing Support Services
M. Knapik, McGraw-Hill
G. Wilkowski, Engineering Mechanics
B. Henry, Fauske & Assoc. Inc.
L. Hendricks, NEI
A. Wyche, Bechtel
D. Raleigh, Scientech
A. Nelson, NEI
J. Rycyna, CWS
D. Miller, Entergy
A. Marion, NEI
L. Mathews, Southern Nuclear
S. Doctor, PNNL

NRC STAFF (July 12, 2001)
T. Gouan, RES
M. Mayfield, RES
R. Elliott, NRR
G. Hubbard, NRR
J. Hopkins, NRR
L. Rossbach, NRR
C. Craig, NRR
J. Zwolinski, NRR
S. Bajwa, NRR
M. Shuaibi, NRR
N. Chokshi, RES
R. Kenneally, RES
D. Diec, NRR
G. Parry, NRR
E. Chow, RES
J. Ridgely, RES
A. Rubin, RES
H. Vandermolen, RES
S. Newberry, RES
E. Connor, NRR
M. Marshall, RES
D. Dorman, RES
K. Karwaski, RES
J. Hannon, RES
J. Tornes, RES
J. Lamb, NRR
J. Vora, RES
J. Boardman, RES
A. Buslik, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
J. Butler, NEI
T. Taminami, Tepco
R. Janati, PA DEP/BRP
P. Negus, GE
D. Rao, LANL
S. Ashbaugh, LANL
A. Wyche, Bechtel
M. Knapik, McGraw-Hill

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

August 20, 2001

REVISED
SCHEDULE AND OUTLINE FOR DISCUSSION
485th ACRS MEETING
SEPTEMBER 5-8, 2001

WEDNESDAY, SEPTEMBER 5, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Vice Chairman (Open)
   - 1.1) Opening statement (MVB/JTL/HJL/SD)
   - 1.2) Items of current interest (MVB/NFD/SD)
   - 1.3) Priorities for preparation of ACRS reports (MVB/JTL/SD)

2) 8:35 - 10:00 A.M. Proposed Resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance" (Open) (SR/AS)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed resolution of GSI-191.

   Representatives of the nuclear industry will provide their views, as appropriate.

10:00 - 10:20 A.M. ***BREAK**

3) 10:20 - 12:00 Noon EPRI Report on Resolution of Generic Letter 96-06 Waterhammer Issues (Open/Closed) (TSK/MME)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff and the Electric Power Research Institute (EPRI) regarding the EPRI Report, TR-113594, "Resolution of Generic Letter 96-06 Waterhammer Issues."

   [Note: A portion of this session may be closed to discuss EPRI proprietary information]

12:00 - 1:00 P.M. ***LUNCH***

4) 1:00 - 1:30 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)
   Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

5) 1:30 - 2:00 P.M. Subcommittee Report (Open) (GBW/PAB)
Report by the Chairman of the Thermal-Hydraulic Phenomena Subcommittee on the results of the meeting held on July 17-18, 2001 at the Oregon State University.

2:00 - 2:30 P.M. ***BREAK***

6) 2:30 - 4:00 P.M. Reactor Oversight Process (Open) (JDS/GEA/MWW)
6.1) Remarks by the Subcommittee Chairman
6.2) Briefing by and discussions with representatives of the NRC staff regarding the use of performance indicators in the reactor oversight process, initial implementation of the significance determination process (SDP), and technical adequacy of the SDP to contribute to the reactor oversight process.

Representatives of the nuclear industry will provide their views, as appropriate.

4:00 - 4:20 P.M. ***BREAK**

7) 4:20 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:
7.1) EPRI Report on Resolution of Generic Letter 96-06 Waterhammer Issues (TSK/MME)
7.2) Reactor Oversight Process (JDS/GEA/MWW)
7.3) Proposed Resolution of GSI-191, Assessment of Debris Accumulation on PWR Sump Pump Performance (SR/AS)

THURSDAY, SEPTEMBER 6, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

8) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GEA/JTL/SD)

9) 8:35 - 9:00 A.M. Peer Review of PRA Certification Process (GEA/MTM)
9.1) Remarks by the Subcommittee Chairman
9.2) Report by Mr. Markley, ACRS Senior Staff Engineer, regarding the application of the PRA certification process described in NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," for the North Anna Power Station that was conducted by the Westinghouse Owners Group and discussed with the licensee on July 16-20, 2001 in Richmond, Virginia.

10) 9:00 - 10:00 A.M. Meeting with the NRC Commissioner Merrifield (Open) (GEA/JTL)
Meeting with Commissioner Merrifield to discuss items of mutual interest.

10:00 - 10:20 A.M. ***BREAK***

11) 10:20 - 12:00 Noon TRACG Best-Estimate Thermal-Hydraulic Code (Open/Closed) (GBW/PAB)

11.1) Remarks by the Subcommittee Chairman

11.2) Briefing by and discussions with representatives of the NRC staff and the GE Nuclear Energy regarding the General Electric TRACG best-estimate code and its application for anticipated operational occurrences transient analyses.

[NOTE: A portion of this session may be closed to discuss General Electric Proprietary Information.]

12:00 - 1:00 P..M. ***LUNCH***

12) 1:00 - 2:00 P.M. Proposed Final Revision to Regulatory Guide 1.78 (DG-1089), "Main Control Room Habitability During a Postulated Hazardous Chemical Release" (Open) (DAP/NFD)

12.1) Remarks by the Subcommittee Chairman

12.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed final revision to Regulatory Guide 1.78.

Representatives of the nuclear industry will provide their view, as appropriate.

2:00 - 2:20 P.M. ***BREAK***

13) 2:20 - 7:00 P.M. Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

13.1) TRACG Best-Estimate Thermal-Hydraulic Code (GBW/PAB)

13.2) Reactor Oversight Process (JDS/GEA/MWW)

13.3) Proposed Resolution of GSI-191, Assessment of Debris Accumulation on PWR Sump Pump Performance (SR/AS)

13.4) EPRI Report on Resolution of Generic Letter 96-06 Waterhammer Issues (TSK/MME)

13.5) Proposed Final Revision to Regulatory Guide 1.78 (DAP/NFD)

FRIDAY, SEPTEMBER 7, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GEA/JTL/SD)

15) 8:35 - 9:30 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

15.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

15.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and

organizational and personnel matters relating to the ACRS.

16) 9:30 - 6:00 P.M. Proposed ACRS Reports (Open)
(12:00-1:00 P.M. LUNCH) Continue discussion of proposed ACRS reports listed under Item 13.

SATURDAY, SEPTEMBER 8, 2001, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

17) 8:30 - 11:30 A.M. Proposed ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 13.

18) 11:30 - 12:00 Noon Miscellaneous (Open) (GEA/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

12:00 Noon **Adjourn**

NOTE:

- **Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.**
- **Number of copies of the presentation materials to be provided to the ACRS - 35.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
484TH ACRS MEETING
JULY 11-13, 2000

[Note: Some documents listed below may have been provided or prepared for Committee use only. These documents must be reviewed prior to release to the public.]

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman
	1. Items of Interest, dated July 11-13, 2001
2	Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50
	2. Risk-Informing 10 CFR 50.46 presentation by M. Drouin and A. Kuritzky, RES [Viewgraphs]
3	SECY-01-0100, "Policy Issues Related to Safeguards, Insurance and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools
	3. NRR presentation, B. Huffman [Viewgraphs]
	4. Industry Views on SFP Risk Study and Policy Options [Viewgraphs]
4	Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"
	5. Letter dated June 26, 2001, from David Lochbaum, Union of Concerned Scientists, to Christopher Grimes, NRR, Subject: Revision to the License Renewal Rule [Handout No. 4-1]
	6. License Renewal Rulemaking Recommendations presentation by NRR [Viewgraphs]
5	Control Rod Drive Mechanism (CRDM) Cracking
	6. NRC Proposed Bulletin to Address: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles presentation by J. Strosnider, NRR [Viewgraphs]
	7. MRP - Alloy 600 ITG RPV Penetrations presentation by NEI [Viewgraphs]
9	Draft Individual Plant Examination of External Events (IPEEE) Insights Report
	8. Perspectives Gained from Individual Plant Examination of External Events (IPEEE) Program presentation by A. Rubin, RES

10 Proposed Resolution of Generic Safety Issues (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance"

9. Results of GSI-191 Parametric Evaluation presentation by RES [Viewgraphs]

11 Potential Margin Reductions Associated with Power Uprates

10. Need for Standard Review Plan Section for Power UprateReviews presentation by NRR [Viewgraphs]

13 Future ACRS Activities/Report of the Planning and Procedures Subcommittee

11. Final Draft Minutes of Planning and Procedures Subcommittee Meeting - July 10, 2001 [Handout #**13.1**]

14 Reconciliation of ACRS Comments and Recommendations

12. Reconciliation of ACRS Comments and Recommendations [Handout #14.1]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Proposed Risk-Informed Revisions to 10 CFR 50.46 and Proposed Revisions to the Framework for Risk-Informing the Technical Requirements of 10 CFR Part 50
	1. Table of Contents
	2. Proposed Agenda
	3. Status Report dated July 11, 2001
	4. Staff presentation handouts, ACRS Briefing, June 6, 2001
	5. 10 CFR 50.46, Acceptance criteria for emergency core cooling ststems for light-water power reactors
	6. Letter from A. Pietrangelo, NEI, to T. King, Questions on LBLOCA Redefinition Program, Project Summary, dated January 8, 2001
	7. Letter from R. H. Bryam to T. L. King, "WOG Large Break Loss of Coolant Accident (LBLOCA) Redefinition Discussion of Benefits," dated October 17, 2000
	8. Letter from A. Heymer to M. Drouin, Large Break LOCA Redefinition Program, Project Summary," dated January 8, 2001
	9. Letter from J. F. Colvin to R. A. Meserve, "SECY-99-264, Proposed Staff {;am fpr Risk-Informing Technical Requirements in 10 CFR Part 50," dated January 19, 2000
	10. SRMs dated January 19, 2001 (SECY-00-0198) and February 3, 2000 (SECY-00-264)
3	SECY-01-0100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools"
	11. Table of Contents
	12. Proposed Schedule
	13. Status Report dated July 11, 2001
	14. ACRS Report dated April 13, 2000
	15. ACRS Report dated November 8, 2000
	16. EDO Response dated January 18, 2001
	17. Analysis of EDO Response dated January 25, 2001
	18. Letter from W. Travers (EDO) to the Commission (12/20/00)
	19. Letter from R. Beedle (NEI) to S. Collins (NRR) (4/26/01)
	20. SECY-01-0100 dated June 4, 2001
4	Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"
	21. Table of Contents

22. Proposed Agenda
23. Status Report dated July 11, 2001
24. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"
25. Memorandum dated June 4, 2001, from Douglas J. Walters, Nuclear Energy Institute, to Christopher Grimes, NRR, Subject" "License Renewal Rulemaking."

5 Control Rod Drive Mechanism Cracking

26. Table of Contents
27. Proposed Schedule
28. Status Report dated July 11, 2001
29. Draft NRC Bulletin 2001-XX: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles
30. Briefing slides, presentation made to the Industry on the Status of the Development of Generic Communication on Vessel Head Penetration Nozzle Cracking and MRP response to NRC RAIs, by A. Hiser, NRR, dated July 3, 2001
31. Briefing slides, presentation made to CRGR, NRC Proposed Bulletin to address Circumferential Cracking of Reactor Vessel Head Penetration Nozzles, A. Hiser, NRR, dated July 2, 2001
32. Memorandum dated June 21, 2001, from C. E. Carpenter, NRR, to W. Bateman, NRR, Subject: Summary of June 7, 2001, meeting with the EPRI Materials Reliability Program on Generic Activities Related to CRDM Cracking
33. Timeline for Control Rod Drive Mechanism Cracking Issue Generic Communication Bulletin

9 Draft Individual Plant Examination of External Events (IPEEE) Insights Report

34. Table of Contents
35. Proposed Schedule
36. Status Report dated July 12, 2001
37. US Nuclear Regulatory Commission, NRC Generic Letter No. 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities-10 CFR 50.54(f), dated June 28, 2001
38. US Nuclear Regulatory Commission, NRC Generic Letter No. 88-20, Supplement 5, Individual Plant Examination of External Events for Severe Accident Vulnerabilities-10 CFR 50.54(f), dated September 8, 1995
39. ACRS Report dated March 8, 1996, Subject: Use of Individual Plant Examinations in the Regulatory Process
40. ACRS Report dated March 8, 1996, Subject: Potential Use of IPE/IPEEE Results to Compare the Risk of the Current Population of Plants with the Safety Goals

41. Related ACRS Letters/Reports on Generic/Unresolved Safety Issues:
    - GSI-148, Smoke Control and Manual Fire Fighting, November 12, 1999
    - Priority Ranking of GSIs: Tenth Group, October 16, 1998
    - SECY-98-001, Mechanism for Addressing GSIs, March 16, 1998
    - Resolution of Multiple System Response Program Issues, June 3, 1996

10 Status of the Proposed Resolution of Generic Safety Issue-191, "Assessment of Debris on PWR Sump Pump Performance"

42. Table of Contents
43. Proposed Schedule
44. Status Report dated July 12, 2001
45. Executive Summary of the Draft Technical Letter Report, "Potential for Loss of NPSH Due to LOCA Debris Accumulation on ECCS Suction Screens" dated June 2001, Revision 0
46. Staff's Handouts for the ACRS Presentation

11 Potential Margin Reductions Associated with Power Uprates

47. Table of Contents
48. Presentation Schedule
49. Project Status Report dated July 12, 2001
50. Memorandum to G. Wallis, Chairman, T/H Phenomena Subcommittee, from P. Boehnert, ACRS Staff, Subject: Working Copy of Minutes of June 12, 2001 Thermal-Hydraulic Phenomena Subcommittee Meeting, dated June 25, 2001 **(Internal Committee Use)**
51. Memorandum to G. Wallis, ACRS, from V. Shrock, ACRS, Consultant, Subject: T/H Subcommittee Meeting, June 12, 2001, Review of GE's Extended Power Uprate Program **(Internal Committee Use)**
52. "Status Report for ACRS Presentation at June 6-8, 2001 Meeting - Potential Margin Reductions for Re-Licensed/Updated Nuclear Power Plants," by A. W. Cronenberg, ACRS Senior Fellow, undated **(Internal Committee Use)**
53. Meeting Handouts, "Signature Estimates of Margin Reductions for Power Uprates/License Renewal," A. W. Cronenberg, ACRS Senior Fellow, Presented at June 12, 2001 T/H Phenomena Subcommittee Meeting
54. Meeting Handouts, "Review of Power Uprates and Potential Synergistic Safety Issues," A. W. Cronenberg, ACRS Senior Fellow, Presentated at June 12, 2001 T/H Phenomena Subcommittee Meeting.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

484TH FULL COMMITTEE MEETING

JULY 11-13, 2001

JULY 11, 2001
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Allen Hiser	NRR/DE/EMCB
KEITH VICHMAN	NRR/DE/EMCB
Mary Drouin	RES/PRAB
Alan Kuritzky	RES/PRAB
WILLIAM BECKNER	NRR/DRIP
Isabelle Schoenfeld	OEDO
Samuel Lee	NRR/DRIP/RGEB
Alan Levin	OCM/RAM
Bill Huffman	NRR/DRIP
RL Sullivan	NRR/DIPM
George Hubbard	NRR/SPLB
Phillip Ray	NRR/DRIP
Tim Collins	NRR/DSSA
Rob Temps	SFPO/NMSS
Phil Brochman	SFPO/NMSS
Steven West	NRR
BT MENDELSOHN	NMSS/FCSS
MICHAEL WEBER	NMSS/FCSS
LARRY RITIGLIO	NMSS/DWM
Matt Blevins	NMSS/DWM
Vonna Ordaz	NRR/DIPM
BARRY ZALCMAN	NRR/DRIP/RGEB
WAN C LIU	NRR/DRIP/RLSB
ROB ELLIOTT	NRR/DSSA/SPLB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

484TH FULL COMMITTEE MEETING

JULY 11-13, 2001

JULY 11, 2001
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Bill Bateman	NRR/DE/EMCB
IRA DINITZ	NRR/DRIP
Barry Marili	NRR/DIPM
BOB SKELTON	NRR/DIPM
Bonnie Schnetzler	NRR/DIPM
BOB PALLA	NRR/DSSA
Frank Gillespie	NRR/DRIP
GOUTAM Bagchi	NRR/DE
Kathy Halvey Gibson	NRR/EPHP
Jennifer White	RES/DSARE
Greg Galletti	NRR/DIPM
Jit VORA	RES/DET/MEB.
Kimberley Rico	NRR/DRIP/RLSB
BRIAN THOMAS	NRR/DSSA/SPLB
JOSEPH E. CARRASCO	NRR/DRIP/RLSB
SAM LEE	NRR/DRIP/RLSB
P T KUO	NRR/DRIP/RLSB
Chris Grimes	NRR/DRIP/RLSB
Ram J. KANGA	NRR/DRIP/RLSB
Jim STRNISHA	NRR/DRIP/RLSB
Steve Hoffman	NRR/DRIP/RLSB
S.K. MITRA	NRR/DRIP/RLSB
Steve Koenick	NRR/DRIP/RLSB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
484TH FULL COMMITTEE MEETING

JULY 11-13, 2001

JULY 11, 2001
TODAY'S DATE

ATTENDEES - PLEASE SIGN BELOW

PLEASE PRINT

NAME	AFFILIATION
Lewis A. Ward	WOG / Southern Nuclear
Robin Graybeal	Protopower
Paige Neyus	GE
Rich Janati	PA DEP / BRP
Roger Huston	Licensing Support Services
MIKE KNAPIK	McGRAW-HILL
Gery M. Wilkowski	Engineering Mechanics Corp. of Columbus
BOB HENRY	FAUSKE & ASSOC. INC.
Lynnette Hendricks	NEI
Brian McCabe	NRC/OCM/JSM
Raul Hernandez	NRC/NRR/DRIP
Althia Wyche	SERCH Licensing/Bechtel
Dan O'Neal	NRC/DSSA
Deann Raleigh	LIS Scientech
Alan Nelson	NEI
~~J. White~~	
J. RYCYNIA	CNS
D. Bryan Miller	Entergy
A MARION	NEI
Rani Franovich	NRR/DRIP/RLSB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
484TH FULL COMMITTEE MEETING

JULY 11-13, 2001

JULY 11, 2001
TODAY'S DATE

ATTENDEES - PLEASE SIGN BELOW

PLEASE PRINT

NAME	AFFILIATION
Larry Mathews	Southern Nuclear
D Bryan Miller	Entergy
Steven Doctor	PNNL

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

484TH FULL COMMITTEE MEETING

JULY 11-13, 2001

JULY 12, 2001
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Tekia Govan	RES
Michael Mayfield	RES
Rob Elliott	NRR/DSSA/SPLB
George Hubbard	NRR/SPLB
Jon Hopkins	NRR/DLPM
Larry Rossbach	NRR/DLPM
Claudia Craig	NRR/DLPM
John Zwolinski	"
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

484TH FULL COMMITTEE MEETING

JULY 11-13, 2001

JULY 12, 2001
Today's Date

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
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1

ITEMS OF INTEREST

484th ACRS MEETING

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
484th MEETING
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Page

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Research: the Vision and Needs of Regulators

REMARKS OF

Dr. Richard A. Meserve
Chairman, U.S. Nuclear Regulatory Commission

at the

Organization For Economic Cooperation And Development's Nuclear Energy Agency (OECD/NEA) Joint Committee On Nuclear Regulatory Activities/Committee On The Safety Of Nuclear Installations (CNRA/CSNI) Workshop

Paris, France
June 19, 2001

I am pleased to participate today in this discussion of research. As it happens, this is an issue that has received very careful and recent attention in the U.S. The Nuclear Regulatory Commission (NRC) has recently received a thoughtful report on the research program by a panel chaired by former Commissioner Rogers and has benefitted by an analysis of NRC research activities by the Advisory Committee on Reactor Safeguards. This meeting is timely because we are considering these reports very carefully in the budget process that is now underway.

I will focus my comments on three questions: Why should a regulator support research? What types of research should be undertaken? And what is the role for international cooperation in research?

Why Support Research?

Some in this audience may wonder why this question should even be asked. As it happens, some licensees have raised questions about the need for NRC-sponsored research. This is a natural and appropriate question, in my view, because our licensees are required to pay most of the costs of NRC activities through fees and they have a legitimate interest in assuring that funds are appropriately expended.

The fact that the question has been asked, however, reflects a failure by the agency to explain adequately the nature of and rationale for NRC-sponsored research. Some research sponsored by the Government is performed in order to answer fundamental scientific questions that have no immediately obvious practical application. NRC research is distinctly not of this kind. Rather, NRC-sponsored research is aimed at providing comprehensive knowledge or understanding to meet a recognized or anticipated need: application of the knowledge to a practical problem is the justification for the work. More specifically, the practical problems that are addressed by the NRC's research program relate to the need to develop and maintain a solid technical foundation for the NRC's regulatory policies.

The value of this work is perhaps best revealed by arraying the contributions from past NRC-sponsored research. As you may know, a major emphasis of the NRC in recent years has been the application of risk insights to improve our regulatory program. This effort holds the promise of both improving safety and reducing needless regulatory burden. The underpinnings for the effort is the tool of probabilistic risk assessment -- a tool that has had significant development as a research project by the NRC and its predecessor, the Atomic Energy Commission, and now provides the foundation for the analysis of reactor safety around the world. Other examples of important past research include the studies on nuclear plant aging, which have helped form the technical basis for the NRC's license renewal efforts; development of a new and more realistic source term, which is protective of public health and safety and also reduces regulatory burden; and the NRC's thermal-hydraulics research program, which has developed computer codes such as RELAP and TRAC that are widely used around the world for reactor safety analyses.

My fundamental point is that virtually every major new initiative that the agency has undertaken over the past few years -- license renewal, risk-informed regulation, design certification of advanced reactor designs, assessment of digital instrumentation and control systems, steam generator tube integrity programs, the new source term, and many other examples--has required technical guidance from our research program. And issues now upon the agency demonstrate the need for further related research. For example, research is needed to respond to industry interest in new types of reactors (such as the helium-cooled reactors), to handle applications for higher burnup of fuel and for power uprates, and to deal with materials issues associated with extended terms of reactor operation. In short, an active research program is a fundamental need for the agency.

What Types of Research Should Be Funded?

As I have already indicated, NRC research is launched in order to meet a known or anticipated regulatory need. There are two subcategories of research that require separate consideration: confirmatory research and anticipatory research. Confirmatory research enables the agency to respond to license applications that are now before the agency or that are anticipated to come before the agency in the future-usually the near future. This type of research supports the NRC's front-line regulatory activities and is usually conducted at the request of the offices that are directly responsible for regulatory oversight-our Offices of Nuclear Reactor Regulation and Nuclear Materials Safety and Safeguards. Criteria for conducting confirmatory research include, for example, the need for independent, confirmatory information on safety issues involving fundamental or crucial barriers, such as fuel or fuel cladding, the absence of independent, confirmatory information on new technology or new designs, and the degree of uncertainty in our knowledge.

The NRC also conducts research programs that are more in the distance, research related to evolving technologies or issues that may become important regulatory concerns in the future. Some of this work may also be confirmatory in nature, providing independent assessment of information developed by the nuclear industry, but much of it is what we refer to as "anticipatory" research. These types of programs may not have been requested by our regulatory offices. Rather, this work arises from the examination of industry trends and an effort to try to foresee where the NRC may need information to respond to future regulatory issues. If we wait until these potential issues become actual regulatory concerns, it may be too late to develop the technical information to respond to them in a timely fashion. Thus the need for forward-looking programs.

I should note that the usefulness of anticipatory research may not become apparent for many years after the initiation of the research. A case in point is the NRC's work on probabilistic risk assessment. Work in this area actually predated the NRC - it was initiated in the early 1970s by the old Atomic Energy Commission and was taken over by the NRC when the agency came into being in 1975. Although the NRC gradually increased its use of PRAs in its regulatory activities, it was not until the mid-1990s, more than 20 years after the initial research, that the NRC embarked on a comprehensive effort to risk-inform elements of our regulatory processes. A similar example is the agency's work on pressurized thermal shock, which was conducted long before the program offices were aware of the regulatory need. Although support for anticipatory research may occasionally lead to blind alleys, a thoughtful approach to research planning for the long term is likely to result in benefits that far outweigh the costs.

The challenge is to maintain an appropriate balance between confirmatory and anticipatory research. It is easy to allow anticipatory research to diminish, particularly in a time of declining budgets, because it is easier to justify the need for confirmatory research. But, as my examples have indicated, the failure of the research organization to look over the horizon to prepare for problems that are not yet apparent to the program offices is a critical need for a regulator.

I should add that the recent evaluation by the outside panel reached the conclusion that the allocation of funds for anticipatory research at the NRC has grown too small. This is an issue that we will need to examine in our ongoing budget review.

What Is the Role for International Cooperation in Research?

It is my view that the NRC must seek international cooperation in research for several reasons.

One reason arises from the harsh reality of budget stringency. The NRC's research budget has declined from over $200 million in the early 1980s to just $40 million in FY 2001, before adjustment for inflation. We are aware that other countries have suffered similar reductions on their programs. There thus is a continuing value in leveraging funds by collaboration on research programs in which there is bilateral or multilateral interest. All participants in such programs benefit from the pooling of resources and the realization of greater efficiencies. The value to each individual participant is much greater than that party's contribution.

Saving money is not the sole purpose for conducting international cooperative research. We recognize that many of our research partners have unique facilities. International collaboration provides broader access to such capabilities. One need only look at the international scope of the testing and analysis that was carried out on reactor thermal-hydraulics in facilities of different scales and capabilities throughout the world -- the Large Scale Test Facility in Japan, BETHSY in France, SPES in Italy, and Semiscale and LOFT in the U.S. The computational tools that are available today for reactor safety analyses are based on data from

these and other test facilities too numerous to list. And the diversity of these testing and analysis programs is also a significant advantage because it promotes both depth and breadth in the available research results.

Another benefit of international cooperation and collaboration is the magnification of intellectual firepower that comes from interactions among a broad set of researchers. The ability to learn from each other and to bring those new insights to bear on issues of reactor safety is invaluable. International cooperation thus improves our understanding of reactor safety issues and contributes to better reactor safety performance, thereby strengthening us all.

Conclusion

Let me conclude simply by noting that I view a strong research program as a central feature of a sound regulatory system. There are challenges in sustaining such activity, particularly in the need to maintain an appropriate balance between confirmatory and anticipatory research. The enhancement of international cooperation in research is essential. Thank you.

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"THE EVOLUTION OF SAFETY GOALS AND THEIR CONNECTION TO SAFETY CULTURE"

by

Dr. Richard A. Meserve, Chairman
United States Nuclear Regulatory Commission
Atomic Energy Society Of Japan/American Nuclear Society Topical Meeting On Safety Goals And Safety Culture

Milwaukee, Wisconsin
June 18, 2001

Introduction

Good afternoon. As General Co-Chairman of this AESJ/ANS topical meeting on Safety Goals and Safety Culture, I would like to add my welcome to Milwaukee. As many of you know, Milwaukee is famous for its beer. It is clear to me that it should also be famous for its hospitality.

The aim of this conference is to explore the development of safety goals, the establishment of a safety culture, and the ways in which these two concepts intersect. We benefit from broad international participation in this meeting and my expectation is that we will find many common elements in the ways in which different countries approach these issues. But I am sure that we will also see some differences that are worthy of discussion as well.

Safety goals and safety culture may appear to be two largely independent topics. The first refers to objectives established by a regulatory agency to define its regulatory philosophy and approach to the

consideration of risk - especially, the concept of acceptable risk. Safety culture also reflects an element of regulatory philosophy, but can encompass a broader range of issues. We speak, for example, to the need for nuclear plant licensees to establish a culture to promote the safe operation of nuclear power stations. The U.S. Nuclear Regulatory Commission also refers to "safety culture" in discussing the way in which its own staff deals with safety issues. Although there are clearly aspects of safety goals and safety culture that do not bear on one another, the two subjects do have a relationship: the way in which safety goals influence regulatory activities can have an impact on the development and maintenance of an appropriate safety culture.

Let me take a few minutes in this opening session to discuss my views on these subjects from the perspective of the U.S. NRC. I will provide a brief historical perspective on the development of the NRC's safety goals and will discuss the practical implications of applying the safety goals to regulatory activities. I will then describe our perception of safety culture. I will conclude by discussing the intersection between safety goals and safety culture.

Safety Goals

The development of the NRC's safety goals will be discussed at length in tomorrow morning's session. For now, let me provide an overview of the development of the goals, the ways in which they have influenced the NRC's regulatory activities, and some of the challenges that remain in front of us.

The NRC's safety goals are described in our Safety Goal Policy Statement, which was released in August 1986.[1] The development of the Policy Statement began not long after the Three Mile Island accident, and was a first attempt by the Commission to come explicitly to grips with the integration of the quantitative assessment of risk into the regulatory system. A few years earlier, the NRC had funded the Reactor Safety Study, known as WASH-1400 and perhaps even better known as the Rasmussen study. That study represented the first use of probabilistic techniques to estimate the frequency of accidents and their ultimate consequences, thereby allowing a quantitative estimate of risk. The primary issue for the NRC in developing safety goals was to use these techniques to help articulate a level of acceptable risk -- in other words, to define "how safe is safe enough."

The Commission established two goals that are stated in terms of public health risk -- one addressing individual risk and the other addressing societal risk. The risk to an individual is based on the potential for death resulting directly from a reactor accident - that is, a prompt fatality. The societal risk is stated in terms of nuclear power plant operations, as opposed to accidents alone, and addresses the long-term impact on those living near the plant. In both cases, the Commission based its acceptable level of risk on a comparison with other types of risk encountered by individuals and by society from other causes, applying the rule that the consequences of nuclear power plant operation should not result in significant additional risks to life and health. The goals were expressed in qualitative terms, perhaps so the philosophy could be understood by all.

In both cases, however, the Commission also expressed the qualitative goals for the safety of nuclear power plants in terms of individual and societal "quantitative health objectives" or "QHOs." These were established at one one-thousandth of the risk arising from other causes presenting the same type of risk.

It is important to note that the QHOs *per se* have never been directly reflected in the NRC's regulations, but were promulgated to provide guidance as to the level of "public protection which nuclear plant designers and operators should strive to achieve." They were also meant to provide guidance to the NRC

staff to use in the regulatory decision-making process. However, the Commission was clear that the safety goals were not meant "to serve as a sole basis for licensing decisions." In fact, the Commission disclaimed an intent to use the goals in making plant-specific regulatory decisions.

While the safety goals provided a metric to address the question of "how safe is safe enough," practical implementation of the Commission's guidance proved to be difficult. This was the result of the large uncertainties involved in calculation of risk in the mathematical sense of probability times consequences. As a result, the NRC staff began looking for other metrics to use as surrogates for the QHOs in regulatory decision-making.

In 1990, the Commission provided additional guidance to the staff regarding the Safety Goals, endorsing surrogate objectives concerning the frequency of core damage accidents and large releases of radioactivity.[2] The numerical value of one-in-ten-thousand for core damage frequency (CDF) was cited as a "very useful subsidiary benchmark...." In addition, a conditional containment failure probability of one-tenth was approved for application to evolutionary light water reactor designs. This resulted in a large release frequency of one in one-hundred-thousand, since containment failure is necessary for a large release to occur. These values have evolved into the "benchmark" values of 10^{4} for CDF and 10^{-5} for large early release frequency (LERF), as discussed in Regulatory Guide 1.174 for use in risk-informed regulatory decision-making.[3]

The application of these goals as an underpinning of the regulatory system has evolved over time from the philosophical to the practical. Now they serve as the basis for many regulatory initiatives. An early example of explicit consideration of risk in a regulation is the NRC's Backfit Rule, originally issued in 1988.[4] But we have moved on to a much more comprehensive application of risk in our regulations, as most in this audience are undoubtedly aware. The aim, of course, is to use risk as the tool for dissecting and reforming our regulatory system so that the NRC focuses on risk-significant activities, thereby both enhancing safety and reducing needless regulatory burden. In implementing this approach we still adhere to many of the basic concepts discussed in the original Safety Goal Policy Statement, such as the use of risk as only one factor among many in making regulatory decisions.

In short, the development of a practical application of the safety goals and the ancillary tool of PRAs have taken many years, but they have growing significance as the foundation for the NRC's work. That being said, there are challenges that must be confronted. Let me mention a few.

First, we recognize that risk, at least for the foreseeable future, will be only one factor that can guide regulatory decisions. In this connection, I want to emphasize the relationship of risk insights to defense in depth. If one had complete confidence in the accuracy of PRAs, one might conclude that defense in depth could be ignored if the risk were sufficiently low. But the Commission is not prepared to jettison the deterministic processes and the defense-in-depth philosophy that are integral parts of the regulatory system. Defense in depth is to be applied at a high level -- that is, to require both prevention and mitigation -- and then as well at lower levels to compensate for uncertainty. There has been much discussion within the NRC and with our Advisory Committee on Reactor Safeguards as to how defense in depth should be incorporated into a risk-informed regulatory approach and this discussion will no doubt continue.

Second, we may need to reconsider the subsidiary objectives. Although the CDF and LERF goals have proven to be quite useful and valuable in implementing the Commission's safety philosophy, they do tend to skew the focus of attention to severe reactor accidents. While it is unquestionably true that the societal risk from nuclear power is dominated by accidents that have low frequencies and high consequences, the perception of risk on the part of the public is influenced by events of low consequence in terms of radioactive releases, but which have much higher frequencies. This is illustrated, for example, by the

reaction following the steam generator tube failure at the Indian Point 2 station in February 2000. The event was widely reported to have involved a release of radioactivity to the environment, although the release was determined to be so slight that the monitoring equipment around the plant could not detect it. Nonetheless, there was an intense public reaction to the event, which continued for several months and ha only recently begun to subside. The safety strategy should address plantoperations, not just accidents, and should consider the full spectrum of events on a frequency/consequence continuum rather than just extreme events. That is, even a low-consequence event is of concern if its frequency of occurrence is high.

Finally, while we wrestle with incorporating risk insights into our current regulatory processes, we face other practical challenges as well. As you know, in the past few months there has strong interest in exploring new construction. We fully expect to see aggressive use of PRAs in connection with new reactor designs as means of satisfying the Commission's goal of assuring that advanced reactor concepts meet or exceed the level of safety provided by the current generation of reactors. Of course, PRAs are now used in the design process itself, to pinpoint and correct vulnerabilities based on risk insights. In this connection, we are grappling with the possibility that we may have to develop a new regulatory system that, unlike the focus of the current rules on light water reactors, will be independent of technology. The foundation of any such system must inevitably include compliance with the safety goals-or their subsidiary objectives-as demonstrated by PRAs.

Despite these many challenges, the NRC is clearly moving in the direction of greater reliance on quantitative tools and goals -- thereby achieving the promise first signaled by the Commission's Safety Goals nearly 15 years ago. I believe the next 15 years will see accelerated progress.

Safety Culture

Let me turn now to safety culture.

Whereas safety goals are relatively straightforward - at least in concept - safety culture is a much broader and, perhaps, less clearly defined concept. There does, however, seem to be general agreement across the industry as to what "safety culture" requires in terms of maintaining superior performance in plant operations. Elements of safety culture include management emphasis on safety as the highest priority; training for all staff, at all levels, to ensure that each employee understands his or her responsibilities for ensuring safe operations; conservative, safety-conscious decisionmaking; a philosophy of continuous improvement, including critical self-assessment and a questioning attitude; and in the event that problems do arise, a willingness to address problems promptly and effectively. Most important, perhaps, is the fostering of a safety-conscious work environment -- one in which plant staff feel they can (and do) raise concerns without fear of adverse consequences. All of these attributes work together to establish a climate that nurtures high safety performance. Safety culture goes right to the heart of the factor that has been shown in research studies to be of paramount importance for excellence in plant operations: human performance.

Just as safety culture is important in nuclear plant operating organizations, the NRC has a responsibility to maintain a strong safety culture among its own staff. Not surprisingly, the elements of safety culture at the NRC are essentially the same as those we expect from our licensees: management involvement, training, conservatism, a questioning attitude, and an atmosphere in which the staff can raise concerns without fear of retribution. And just as our licensees have on occasion had to deal with problems in maintaining a strong safety culture, the NRC has challenges in this regard as well. Nonetheless, when one looks at the vastly improved performance of the industry in terms of both safety and operations, it appears that both the industry and the NRC have had a fair measure of success in fostering a strong safety culture.

One management challenge is to continue maintain an appropriate culture over time; all too often, we have seen operational excellence eroded by complacency. As a result, we must provide continuing emphasis on safety culture; we will continue to stress the need for vigilance both for our licensees and for ourselves.

The Intersection of Safety Goals and Safety Culture

On a fundamental level, safety goals and safety culture are linked together. In a very real sense, safety culture is a significant contributor to the ability to meet safety goals. This is not to suggest that plant operators use the QHOs or the subsidiary objectives as numerical targets on a day-to-day basis, but rather that a strong safety culture leads to an operational philosophy consistent with the safety goal objectives of minimizing risk. And the connection is becoming closer. Some licensees have begun to employ on-line quantitative risk evaluation to assist in making safety-focused operational decisions.

The NRC's safety culture, as manifested in the staff's approach to regulation, can have an impact on a licensee's safety culture. Over-regulation has the potential to rob a licensee of a sense of "ownership" of the safety performance of a plant, which can degrade licensee performance. Under-regulation has its own obvious set of perils. Thus, the NRC's culture must find the appropriate balance in the oversight process so as to maintain an adequate safety focus without creating unwanted impacts on licensee safety culture. One part of that balance is an appreciation of the role of licensee safety culture in the achievement of safety goals. And so the connection between safety culture and safety goals is again revealed.

Concluding Remarks

In these remarks, I hope to have given you a sense of how the NRC views the issues of safety goals and safety culture, and the ways in which these important concepts interact. I believe that a strong safety culture, augmented by an appreciation for the risk implications of actions of both licensee and regulatory organizations, can assist in the development and maintenance of excellence in nuclear plant operational safety.

Thank you.

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1. U.S. NRC, "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," 51 *Federal Register* 30028, August 21, 1986.

2. U.S. NRC, Staff Requirements Memorandum on SECY-89-102, "Implementation of the Safety Goals," June 15, 1990.

3. U.S. NRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, July 1998.

4. Code of Federal Regulations, Title 10, Part 50, §50.109

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Remarks of

Dr. Richard A. Meserve,
Chairman, U.S. Nuclear Regulatory Commission
At the
Sixteenth Annual
Korean Atomic Industrial Forum/Korean Nuclear Society Conference
The Direction of Nuclear Regulatory Policy in the U.S.A.
Seoul, Korea

April 17, 2001

INTRODUCTION

I am pleased to participate in the annual conference of the Korea Atomic Industrial Forum and the Korean Nuclear Society during my first visit to Korea as Chairman of the United States Nuclear Regulator Commission.

In his letter of invitation, Dr. Choi asked me to discuss my perspectives on the direction of nuclear regulatory policy in the U.S. I will thus share some thoughts with you on the U.S. NRC's regulatory activities - where we are now, and where we are headed in the future. I will also address the importance of international cooperation in enhancing our regulatory process, both at home and abroad. Before commenting on these matters, I want to take a moment to reflect upon the remarkable time in which we find ourselves in the United States.

THE ENERGY CONTEXT

The U.S. is experiencing a period of changing attitudes toward nuclear power. Only a few years ago, pundits claimed that the deregulation of the electricity business would result in the premature shutdown of many nuclear plants and the eventual end of reliance on nuclear power in the U.S. In striking contrast to these forecasts, we in fact have seen a renewed interest in nuclear energy. Many licensees seek to extend, rather than shorten, the expected lives of their plants. There also is a strong competition among a variety o bidders to acquire ownership of existing nuclear plants, in recognition of their economical, reliable, and environmentally benign performance. And we have even seen the first stirring of interest in the possibility of new construction -- a thought that would have been unthinkable even a year ago.

An important factor in the emerging attitudes toward nuclear power is the remarkable improvements in nuclear plant performance over the past decade. The average capacity factor for U.S. light water reactors was over 90 percent for the first 9 months of 2000, up from approximately 65 percent just 10 years ago. Performance indicators show that during the same period the overall safety performance of the industry has significantly improved. For example, the average number of automatic scrams has declined by approximately a factor of 3 in the past decade. This improved performance has resulted in significant increases in electrical output; in fact, nuclear electrical output has grown approximately 25 percent in the last decade without the introduction of any new plants. As a result, electricity production from U.S. nuclear plants is now second only to that produced from coal-burning plants.

These changing attitudes have been reinforced by the problems with electrical supply in the State of California. The core problem is rather elementary: there is insufficient generation capacity to meet growing electricity demands. The nuclear plants in the western U.S. are appropriately seen as the anchors of the grid. Even some of those who have opposed nuclear power in the past recognize and value the important contribution of the nuclear sector to electricity supply.

Although deregulation may be slowed in some states in the aftermath of the California situation, the supply problems in the western U.S. have prompted the start of the first careful scrutiny of national energy policy in the past 20 years. The new Administration in Washington has formed a task group chaired by Vice-President Cheney. And there is strong Congressional interest in energy legislation, as reflected in several bills that are already pending. The early discussions suggest that nuclear power will be a strong component in the mix of technologies that are shaped into a national strategy.

The NRC does not have a promotional role for nuclear power in this debate. Indeed, the NRC's fundamental mission and responsibilities remain unaltered. The NRC is obligated to regulate the Nation's civilian use of nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment. Because the viability of the nuclear option is absolutely dependent on the maintenance of safe operations, the NRC's -- and the industry's -- highest priority must be the protection of public health and safety. If we fail in ensuring safety, the emerging optimism about nuclear energy will quickly disappear.

Although the NRC's focus must remain on safety, this does not mean the NRC has no role in the resurgent interest in nuclear power. The NRC's regulatory system should not establishinappropriate impediments to the application of nuclear technology. The NRC's performance goals reflect this philosophy: they include the improvement of the efficiency and effectiveness of our regulatory process and the reduction of unnecessary regulatory burden. Many of our initiatives over the past several years have sought to maintain safety -- our primary performance goal -- while simultaneously simplifying and improving our regulatory system. The NRC also has an important obligation to establish and maintain public confidence -- another c our performance goals. In fact, we believe the NRC fosters a climate in which the nuclear option can be fairly evaluated by both being a strong regulator and by being seen by the public as fulfilling that role.

The role of nuclear energy in the U.S. over the coming decades is dependent on continuing safe operation of our existing fleet and, if society so decides, on new construction. Let me turn my discussion to certain NRC-related activities that bear on these matters.

LICENSE RENEWAL

The limitation in U.S. law to a 40-year term for an initial operating license was not established on the basis of technical limitations, but rather was driven by antitrust and financial considerations. The law allows the NRC to consider a license renewal and we will grant such a renewal if, after a full evaluation, we conclud a plant can be safely operated for an extended period. The first license renewal applications, for Calvert Cliffs and Oconee, were received in 1998, and the NRC developed an ambitious 30-month schedule to complete the safety and environmental evaluation of each application. We met our schedules for both plants and approved 20-year extensions last year. We currently have three applications under review, including the first boiling water reactor, Southern Company's Hatch plant. Five additional applications are expected during the coming fiscal year. Roughly 40 percent of U.S. plants have formally expressed their intention to seek license renewal, and ultimately more than twice that many may apply. These renewal applications, if successful, will mean that nuclear energy will contribute significantly to U.S. energy supply well into this century.

The Commission recognizes that the simultaneous review of many renewal applications presents a considerable challenge in managing resources. But I am confident that we're up to the task. We must -- and shall -- fulfill our responsibilities to perform high-quality, technically sound reviews while maintaining the efficient, effective process that has been established in these first reviews.

CONSTRUCTION OF NEW NUCLEAR POWER PLANTS

Increased demands for electricity in the future will need to be addressed by construction of new generating capacity of some type and, as I have mentioned, serious industry interest in new reactor construction in the U.S. has recently emerged. The Commission, working with current licensees and other stakeholders, has put in place a more efficient licensing procedure to avoid some of the delays incident to the processes under which the current fleet of plants was licensed. In the last few years, the NRC has certified three advanced reactor designs: the General Electric advanced boiling water reactor, the Combustion Engineering System 80+, and the Westinghouse AP600 light water reactors. In addition to these certified designs, there are new nuclear power plant technologies, such as the Pebble Bed Modular Reactor, which some believe can provide enhanced safety, improved efficiency, lower costs, as well as other benefits. Many of these designs are likely to first be built in other countries, and the NRC will be looking to our exchange programs to provide us with operational data which will be used in later licensing decisions. For example, I know that Korea may have early operational experience that bears particularly on the System 80+ design.

To ensure that the NRC is prepared to evaluate any applications to introduce these advanced nuclear reactors, the Commission is assessing its policies to identify where changes may be necessary. Particular emphasis is being placed on the early identification of regulatory issues. Moreover, the staff is assessing its technical, licensing, and inspection capabilities in order to identify enhancements that would be necessary to ensure that the agency can effectively carry out its responsibilities.

In order to confirm the safety of new concepts, the Commission believes that a strong nuclear research program must be maintained. A comprehensive evaluation of the NRC's research activities is underway with assistance from a group of outside experts and from the NRC's Advisory Committee on Reactor Safeguards. With the benefit of these insights, it is my intention for the Commission to take steps to

strengthen our research program over the coming months.

I cannot leave this topic without noting the invaluable work performed in our joint international research programs. With budgets that are inadequate for any single country to perform all phases of investigational and confirmatory research, our ability to engage in focused, global cooperation enables all of us to enhance our nuclear regulatory and safety regimes.

RISK-INFORMING NRC REGULATORY ACTIVITIES

An important NRC initiative relates to the reexamination of the foundations of our regulatory system. Improved probabilistic risk assessment (PRA) techniques combined with over 4 decades of accumulated experience with operating nuclear power reactors have caused us to recognize that some regulations may not serve their intended safety purpose. This situation arises because, when many NRC regulations were initially formulated, the NRC did not yet have much practical experience with commercial reactors. As a result, the Commission generally proceeded very cautiously, relying on conservative engineering judgment and defense in depth. We have learned much in the intervening years and now recognize that some of our regulatory requirements may not be necessary to provide adequate protection of public health and safety. Where that is the case, we should revise or eliminate the requirements. On the other hand, we must be prepared to strengthen our regulatory system where risk considerations reveal the need. We are presently evaluating the technical bases of our main body of requirements and modifying them, as appropriate, to focus on risk-significant issues.

One particularly important activity is our effort to risk-inform our reactor inspection process. This new oversight process uses a combination of objective performance indicators and risk-informed inspections to measure plant performance. The new program also incorporates a simplified PRA to determine the risk significance of inspection findings so that the NRC can focus attention on those matters that are most important.

We are close to completion of the first year of initial industry-wide implementation of this new program and, overall, we find that the new process has been a remarkable success. The process has provided a more objective and understandable evaluation of plant performance, with a focus on operational aspects that are of the highest safety significance. And the new process has also improved public access to assessment information and has reduced unnecessary regulatory burden. Notwithstanding our successes in this area, we recognize that improvements can still be made and we seek to engage all of our stakeholders, including the public, in our self-assessment efforts.

I should note that there has been intense interest internationally in our move to risk-informing our regulations, in part because the trend in other countries has been toward a more prescriptive approach. While this difference is real, I do not believe that the contrast is as stark as it is often portrayed. We are building on a long history of prescriptive regulation in the U.S., not eliminating that knowledge base. We use risk insights to supplement or inform modification of our prescriptive requirements. It is for this reason that we urge other countries which may be considering a move to a risk-informed regulatory system to establish a strong safety foundation on which to build risk-informed approaches. The process of risk-informing regulations is not a means to diminish necessary regulatory oversight; rather, with the appropriate safety basis, it is a way to allow the more effective use of resources.

PROGRESS ON HIGH LEVEL WASTE STORAGE AND DISPOSAL

Solutions for high level waste storage and disposal continue present challenges to the NRC. In the past several years, the NRC has responded to numerous requests to approve cask designs for onsite dry storage

of spent fuel. These actions have provided an interim approach pending implementation of a program for the long-term disposition of spent fuel. We anticipate that the current lack of a final disposal site will result in a large increase in on-site dry storage capacity during this decade.

There currently are two potential alternatives to on-site storage -- centralized interim storage, and disposal in a geologic repository. Delays have been encountered with both alternatives. The staff is currently reviewing an application for an Independent Spent Fuel Storage Installation in the State of Utah. And certain matters also need to be resolved in order to make progress on the proposed deep geologic repository at Yucca Mountain in Nevada. I am cautiously optimistic that the regulatory framework for consideration of a possible repository at Yucca Mountain can be in place within the next several months.

MAINTAINING LONG TERM SUCCESS

I want to spend a few minutes in discussing two areas that affect the long-term success of the NRC. The first is the need to maintain the core competency of the NRC staff. My close exposure to the staff over the 17 months I have been with the Commission has served to deepen my appreciation of the dedication, thoughtfulness, and technical skill of the staff. But I am worried about the future. In some important offices, nearly 25 percent of the staff is eligible to retire today. In fact, the NRC has six times as many staff over the age of 60 as it has staff under the age of 30. And it is becoming increasingly difficult for the NRC to hire personnel with the knowledge, skills and abilities to conduct the safety reviews, licensing, and oversight actions that are essential to our safety mission. The number of individuals with the skills critical to the achievement of our safety mission is rapidly declining in our Nation and our educational system is not replacing them. In response to this important issue, the NRC is now seeking systematically to identify future staffing needs and to develop strategies to address the gaps. I mention this issue because I believe this is an international issue that confronts all of us.

NEED FOR PUBLIC OPENNESS

The second matter of importance is the need for public openness. None of the changes that I have described will serve their intended purpose without public confidence in the NRC and in the industry. As we have seen time and again, the willingness of the regulator and the industry to respond quickly to an incident and to keep the public fully informed has had a dramatic impact on the public's response.

There are segments of our society that are very concerned about the risks -- real and imagined -- that nuclear technology presents to the public health and safety and the environment. Others worry about the need to safeguard nuclear materials so that untoward uses are avoided. And others are worried about the risk attendant to nuclear waste. Many of those holding strong views on such matters may not be technically knowledgeable and cannot engage the agency at the level of technical sophistication with which our staff is most comfortable. If the NRC is to be successful, however, the concerns of the public must be openly acknowledged and directly confronted.

Equally important, there is a procedural imperative to make decisions through processes that are accessible to the public. No matter how careful a job we do, if our work is performed behind a wall of secrecy, the public will not have confidence that the result is fair, objective, honest, or in the public interest. There will always be the corrosive suspicion that decisions made outside the sight of the public serve to protect those favored by the decisions, to conceal dangers, or to cloak imprudent acts.

As a result of these considerations, the Commission has strived to maintain open communication with all its stakeholders and seeks to ensure the full and fair consideration of issues that are brought to our attention, whatever the source. Occasionally this means that our decision processes are slow. But, we

believe that public confidence in any increased reliance on nuclear power will not be achieved unless the NRC engages the concerned public and thereby both acts to ensure safety and is seen to act responsibly fo that purpose.

INTERNATIONAL COOPERATION

I would like to conclude by discussing a goal that we all share -- ensuring nuclear safety. Nuclear technology is now pervasive throughout the globe. Over 400 nuclear power plants are now operating in more than thirty nations, supplying about one-sixth of the world's electricity. In several countries, nuclear power supplies over 70 percent of domestic electricity production. And new nuclear capacity is planned or is being considered in a wide range of nations.

The decision whether to use nuclear power, the determination of the number, size, and location of plants, and the designation of the methods to be applied by plant operators and regulatory agencies to ensure safety and public protection are matters of sovereign concern. But there also is a vital need for international cooperation to ensure that safety is the fundamental consideration in the use of nuclear technology.

The nuclear industry has clearly recognized the need for and value of international cooperation and technical information exchange. Indeed, the nuclear business is now international in nearly every aspect: design, construction, operation and regulation. It is imperative that cooperation continue and expand to promote good safety practices and to discourage poor ones.

I am firmly committed to continuing the U.S. NRC's role in international cooperative exchanges at all levels. NRC staff members participate in many international conferences and on many international working groups. The contributions of our international research partners are essential to the vitality of the NRC's research program. On the Commission level, my fellow colleagues and I have met with many of o counterparts around the world to discuss perspectives on nuclear regulation and ways in which to promote adherence to the highest degree of safety assurance. The NRC's Office of International Programs coordinates technical information exchange agreements with 34 other nations.

I am sure that we can do more. It is through interactions such as those provided by this conference that we each can learn from each other.

CONCLUSION

I hope that my remarks have provided you with a sense of the direction of the NRC's regulatory policies. The assurance of public health and safety undoubtedly remains our foremost obligation. With the renewed interest in nuclear power in the U.S., the achievement of safety will require the NRC to anticipate the challenges and to adapt to them. Our continued success benefits greatly from international cooperation and, thus, I am particularly pleased to have had the opportunity to speak with you today. Thank you.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF PUBLIC AFFAIRS

Office of Public Affairs **Telephone: 301/415-8200**
Washington, DC 20555-001 **E-mail: opa@nrc.gov**
Web Site: http://www.nrc.gov/OPA

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SAFETY: THE FOUNDATION UPON WHICH ECONOMIC VALUE IS BUILT

REMARKS OF

JEFFREY S. MERRIFIELD
COMMISSIONER

AT THE
2001 ANS ANNUAL MEETING

JUNE 18, 2001

Good Morning. Thank you very much for the opportunity to speak to you today. I applaud Mike Sellman and his team for putting together this important annual meeting and for developing a program that should prove to be interesting and informative.

Let me start by saying that the state of the U.S. nuclear industry today is very sound and that the outlook for nuclear power is the brightest its been in several decades. By almost any safety, reliability, or economic performance indicator, the 103 operating nuclear power plants in the U.S. are operating better today than ever before. Our licensees have developed sound maintenance and corrective action programs, improved operator training and performance, made significant process improvements, shortened refueling outages, and as a result, significantly increased both the safety and generation of power in the nuclear fleet. This improved performance has resulted in an increase of generation from the existing fleet equivalent to placing 23 new 1000 megawatt power plants on line. This performance has also set the stage for nuclear power's return to the forefront of the energy debate in the United States.

Over the past several months, Americans have been inundated by news reports that describe a renaissance

that is occurring within the nuclear power industry. As you undoubtably know, just last month, the Bush Administration unveiled its national energy plan which calls for nuclear energy to be "a major component of the United States fuel mix". I understand the enthusiasm within the nuclear industry for new plant construction; however, as an NRC Commissioner, I also understand the significant technical, regulatory, and infrastructure challenges that are raised by the prospect of new plants. For example:

- There is serious consideration being given to the reactivation of construction on WNP-1 and Bellefonte. Should our licensees pursue that route, there are regulatory and technical challenges that will have to be addressed.

- Several of our licensees are actively considering applying for an early site permit in the very near future. Given that Part 52 has never been fully exercised, there is understandable uncertainty about the application of the early site permitting process.
- Should a potential licensee actually make the decision to go forward with construction of a**new** plant in the United States, we will face many challenges associated with:

(1) a combined operating license process that has never been exercised;

(2) a human capital pipeline that will have to be rebuilt after many years of neglect; and,

(3) the industry's reliance on foreign manufacturers for large reactor components and regulatory oversight of those manufacturers.

- Finally, should a potential licensee choose to move forward with a**nadvanced** reactor design like the Pebble Bed Modular Reactor, the NRC and the industry will have to meet formidable challenges associated with:

(1) a regulatory infrastructure built around light water reactor technology;

(2) a workforce with limited experience and expertise in these technologies; and,

(3) policy issues pertaining to such things as emergency planning and containment, that will undoubtably have significant public confidence ramifications.

There should be no doubt in anyone's mind that these challenges are real, and they are significant. They will often put both the NRC and the nuclear industry in unchartered regulatory waters. I assure you that I and my fellow Commissioners recognize these challenges and have taken the proactive steps we believe are necessary to ensure the NRC is prepared to carry out its regulatory responsibilities in an effective and efficient manner.

I applaud the ANS for choosing as its theme for this year's annual meeting**Safety Culture and Its Relationship to Economic Value in a Competitive Market**. It is a tribute to the maturity of this industry that despite the success the industry is enjoying and the exuberance over new plant construction, the primary focus of this meeting is<u>Safety</u>. As is so accurately reflected in the program for this meeting, "safety and safety culture are the foundation for the future growth of this industry". I believe that the future of the nuclear industry <u>does not</u> hinge on corporate decisions about new plants -- it hinges on the safety of the existing fleet of reactors. Thus, neither the NRC nor the industry can allow the headlines about new plants to distract us from maintaining the safety of the current fleet, nor can we permit ourselves to lose momentum on the important regulatory improvement initiatives that are underway.

Today, I am going to dissect the two cornerstones of this meeting: Economic Value and Safety Culture. From my perspective, safety and economic value are not only compatible, they're inseparable. Safety is simply the foundation upon which a plant's economic value is built. Anyone who believes that safety and economic value are mutually exclusive goals is simply blind to the realities that history has unmistakably, and sometimes painfully, taught this industry. **Poor safety performance ultimately manifests itself in poor plant reliability and poor economic performance.** Poor safety performance will bring with it severe regulatory consequences and poor plant reliability will undoubtably bring with it the severe economic consequences of a competitive electric market. I will begin today by briefly discussing two important initiatives that should, if done responsibly, maintain safety while significantly enhancing the economic value of plants. I will then share my views on what I believe are three fundamental threats to a plant's safety culture. Finally, I will discuss the economic value of public confidence.

ECONOMIC VALUE

The relationship between economic value and safety is not new to the NRC. In fact, it is at the center of two of the most significant regulatory challenges the NRC faces today: license renewal and power uprates.

License Renewal

License renewal is clearly at the forefront of the industry's efforts to enhance the economic value of its plants. Nuclear power's favorable environmental and economic position relative to fossil plants, the growing need for electric generation in the U.S., and a much more stable and disciplined regulatory environment, have fueled remarkable interest in license renewal. In a speech to the Nuclear Energy Assembly last month, Joe Colvin, NEI's President and CEO, indicated that "renewing the licenses of nuclear plants made enormous economic sense" and that virtually all plants will ultimately seek license renewal. This speaks volumes about the renewed economic value of these plants.

Last year, the NRC renewed the Calvert Cliffs and Oconee licenses for another 20 years. We are well along in our reviews of the renewal applications for ANO, Hatch, and Turkey Point. Just a few weeks ago, we received the applications for North Anna and Surry, and just last week, we received the applications for Catawba and McGuire. On the immediate horizon lies the license renewal application for the two reactors at Peach Bottom. For the NRC, the addition of these 10 reactors to our license renewal process in just a 2-month period represents a challenge -- a daunting challenge -- but a challenge that I am confident we are ready to meet.

My message to licensees considering license renewal is that the recipe for success is quite clear: develop sound programs for managing plant aging, submit renewal applications that are of the highest quality, and ensure that license renewal does not distract your staff from maintaining the operational performance and safety of your plants. My message to all of our stakeholders is that the NRC will never allow safety to be compromised in order to enhance a plant's economic value. We have an obligation to review license renewal applications; we do not have an obligation to approve them. Having said that, I believe that we also have an obligation to ensure that our review process is conducted in as efficient and timely manner as possible. We must plan and budget our resources carefully. We must apply the lessons we have learned from the initial applications so that further process improvements can be made. Finally, we must continue to improve the Generic Aging Lessons Learned and the Standard Review Plan so that future reviews are carried out in a disciplined, consistent, and even more timely manner. In essence, this is the NRC's recipe for success.

Just two years ago, there was considerable uncertainty about whether the NRC could meet its goal of a 36-month review process. Despite this uncertainty, at the 1999 ANS Annual Meeting in Boston, I

challenged the NRC staff to make the process improvements necessary to responsibly achieve an 18-month review schedule. At that time, many individuals within the NRC, and quite frankly, within the industry, felt that I was being unrealistic. Today, the NRC stands on the verge of renewing the ANO-1 license in just 17 months. I am very proud of the fine job our staff has done on the initial license renewal reviews and I applaud them for rising to my challenge. However, if our licensees continue to proceed responsibly, and if the NRC continues to strive to improve the efficiency and effectiveness of its review process, I believe it is not unreasonable to expect that two years from now, the Commission itself may not be satisfied with even an 18-month review process.

Power Uprates

Another initiative that is taking on rapidly growing relevance in the industry's efforts to enhance the economic value of its plants is power uprates. This increased relevance is a result of the economic reality that power uprates are the least costly means by which to increase generation. To date, the NRC has approved approximately 2000 megawatts-electric of power uprates, and has done so in a manner that is protective of public health and safety. Until recently, these uprates were typically on the order of two to seven percent and because licensee interest was somewhat measured, these uprates did not significantly challenge NRC resources. Now, the economics of nuclear power has changed so dramatically, that the NRC finds itself facing an ominous licensing challenge in this area. Many licensees are taking advantage a rule change the NRC made last year to Part 50, Appendix K, and are pursuing power uprates of around 1½ percent. Several BWRs are also capitalizing on a GE Topical Report and have submitted applications for extended power uprates of 15 to 20 percent. Based on information provided to us by the industry, we anticipate that most BWRs will ultimately follow this path. Some industry analysts are predicting that licensees will pursue power uprates totaling 8,000 to 12,000 megawatts in the coming years.

I encourage industry leaders to proceed responsibly in this area. In your quest to get more value from your generating assets, don't jeopardize their future. You must ensure that engineering analyses are sound, safety margins are well understood, and plant reliability is not challenged. You must reinforce to your staff that your corporate commitment to safety must serve as the foundation for any effort to improve the economic value of your plants. Anything short of this amounts to false economics.

As for the NRC, I believe our record demonstrates that we are prepared to review uprate applications in a manner that is fully protective of public health and safety. However, I do not believe our record demonstrates quite so clearly that we can consistently carry out these reviews in a disciplined and timely manner. For example, I am not satisfied with the timeliness and discipline of our reviews associated with the 1½ percent uprates that I just mentioned. The staff recently informed the Commission that it is spending more time and resources reviewing these small uprates than it is on uprates of five percent. This is simply not a risk-informed way of doing business. It is clear to me that process improvements and increased management oversight are absolutely essential to ensure we consistently meet our growing regulatory responsibilities in an efficient and effective manner. While safety is our highest priority, we have a responsibility to the American people to carry out our safety mission in a risk-informed manner that does not inappropriately detract from the economic value of these plants.

SAFETY CULTURE

Let me now turn to the second cornerstone of this conference, **safety culture**, and share my views on what I believe to be three fundamental threats to a plant's safety culture: an ineffective corrective action program, complacency, and insularity.

Corrective Action Programs

I believe one of the greatest threats to a plant's safety culture is an ineffective corrective action program. I challenge anyone to dispute my assertion that the dramatic improvements made in both the safety and economic performance of this industry would not have been possible without the equally dramatic improvements made to plant corrective action programs. Record capacity factors, breaker-to-breaker runs, high levels of equipment reliability, and fewer plant transients do not happen by accident. They happen only when plant management fosters a safety culture which encourages workers to identify problems and finds workarounds intolerable. They happen only when management holds itself accountable for prompt and effective resolution of identified problems. They happen only when management places a high priority on pursuing latent conditions that lie dormant but are poised to reveal themselves during the worst of situations.

Despite the industry's remarkable improvements in this area, corrective action programs at some plants warrant additional attention. To those plants I say, "let history be your guide". The fact is, the history of this industry is marred with plants that have paid a heavy price because management failed in its responsibility to foster a robust corrective action program. These plants paid a staggering price to rectify poor safety and economic performance. However, **that** price pales in comparison to the price paid to correct the resulting unhealthy work environment - an environment in which employees stopped looking for problems and management became tolerant of mediocrity. The NRC believes that effective corrective action programs are so essential to safety that they are a centerpiece of the NRC's new reactor oversight process. Should the NRC staff lose confidence that a licensee's program is robust enough to maintain plant safety, I assure you our regulatory response will be swift and it will be severe. I hope none of our stakeholders expect any less. Also, given that a poor corrective action program will undoubtably manifest itself in a plant's capacity factor and reliability, I would expect that the competitive market will be an equally swift regulator. There's a saying that goes, "If you're not finding problems, you are missing opportunities for growth". I encourage the industry to continue to challenge its corrective action programs to ensure that opportunities for growth are not lost.

Complacency

Another threat to a plant's safety culture is complacency. The nuclear industry must continue to challenge itself to resist the insidious build up of complacency that can occur when organizations become content with their own success. As I have reiterated on many occasions, in the increasingly dynamic environment facing the nuclear industry, those that are content with the status quo will undoubtably become faint images in the rear view mirrors of those that recognize that success must be redefined every time they think they have achieved it. While the industry is performing very well, it was not long ago that many plants were plagued with operational problems. We cannot allow ourselves to forget about the Davis-Besse feedwater event, the fire at Browns Ferry, the Millstone saga, and the extended shutdowns of the 80s and 90s. We cannot allow ourselves to lose sight of the fact that the performance improvements the industry is enjoying today came at a very high price--a price the industry cannot afford to repeat. While recent news coverage centers around the revival of the nuclear industry in the U.S., let's not forget that just five years ago, this industry was on the cover of Time magazine for much different reasons. As they say in Hollywood, do not allow yourself to be seduced by favorable reviews. Complacency is simply this industry's worst enemy--a significant threat to both a plant's safety as well as its economic value.

Insularity

Finally, I believe that insularity is a growing threat to the safety culture of the nuclear industry. I recently read a speech that Mike Sellman gave at the ICONE-9 conference in Nice, France. In that speech, Mike insightfully pointed out that there are no "local mistakes" in this business. I couldn't agree more. I also

believe that there should be no "local solutions" in this business either. As consolidation in the ownership of nuclear plants continues, the few large companies operating these plants must not become insular. They must continue to recognize the value of looking outside their organization for solutions, and of sharing information outside of their organization for the common good of the industry. Plant managers within these large companies must never become comfortable benchmarking themselves only against their organizational peers, mistakenly believing that the rest of the U.S. nuclear fleet and the international community offer few operational insights that cannot be more readily acquired from within. As I have said on many occasions, for those who are so bold as to believe that all of the nuclear industry's solutions, all of its best practices, and all of its operating experience, lie within your organization, I ask you this: "Are you bold enough to stake your assets on it?" I hope and expect the answer is no.

THE ECONOMIC VALUE OF PUBLIC CONFIDENCE

Now, let me turn to an area of great importance to the NRC and the nuclear industry; the issue of public confidence. I applaud the ANS for recognizing in its program that public support for new nuclear construction will only come if there is strong public confidence in the safety of nuclear power and the industry's ability to operate plants responsibly. I couldn't agree more. The resurgence in public confidence that nuclear power is enjoying would not have been possible were it not for the industry's improved safety performance over the last few years. Nonetheless, this confidence is fragile and thus the industry must always be vigilant in protecting it. The best way to do that is by continuing to operate the plants safely, reliably, and efficiently.

I find it very intriguing how the nuclear industry approaches public confidence in such a diverse manner. Some licensees, like Progress Energy, view public confidence and effective public communication as high corporate priorities -- priorities that I believe make good business sense. These licensees understand the economic, social, and political benefits associated with public confidence, and they seize opportunities to enhance it. These licensees recognize that public confidence must be earned and it must be vigilantly protected. Other licensees simply ignore public confidence, seemingly unwilling to spend the time and resources necessary to enhance it. Licensees that adopt this approach do so for a variety of reasons ranging from a mistaken perception that public confidence has no economic value, to a hopeless resignation that public confidence simply cannot be influenced, to a misguided perception that good plant performance speaks for itself and thus public outreach is unnecessary. Finally, there are still a few licensees that recognize the importance of public confidence, but simply do not maintain plant performance at a level that engenders a high degree of it.

My views on this matter are quite clear. Enhancing public confidence and communicating honestly and effectively with the public are not this industry's burdens; they are its responsibilities. I believe that those who dismiss the value of public confidence serve to erode the foundation upon which the future of nuclear power will be built. To those licensees whose plant performance does not engender public confidence, I say fix your problems and fix them expeditiously. Your performance not only undermines public confidence in your plant, but it has the spillover effect of eroding public confidence in each of the 103 reactors operating throughout the U.S. To those licensees who believe public confidence has no economic value, I encourage you to try to make that argument to your colleagues at Indian Point 2. I am quite certain that ConEd found the economic burdens associated with facing a public that had lost confidence in their ability to operate the plant safely to be quite severe. Finally, to those licensees that mistakenly believe that public confidence cannot be enhanced, I encourage you to learn from your colleagues at Millstone, who were once paralyzed by a complete loss of public confidence, but who have made significant strides in the difficult and costly journey of earning this confidence back.

In sum, it is indeed difficult to quantify the economic value of public confidence. However, as those plants

that have lost it can attest, the economic impacts associated with restoring lost public confidence are real, they are quantifiable, and they can be staggering.

CONCLUSION

In closing, William Jennings Bryan once said, "Destiny is not a matter of chance; it's a matter of choice. It is not a thing to be waited for; it is a thing to be achieved." The destiny of the nuclear industry wilhot be defined by corporate decisions surrounding new plant construction. Instead, it will be defined by those men and women responsible for operating and maintaining the existing nuclear fleet, and by those industry leaders who are ultimately responsible for fostering a healthy safety culture within their organizations. The stakes are high and the burdens great. However, if recent performance is any indication, I am confident that the industry is up to the challenge and is fully committed to ensuring that its destiny is not left to chance. Thank you very much.

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U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs **Telephone: 301/415-8200**

Washington, DC 20555-001 **E-mail: opa@nrc.gov**

Web Site: http://www.nrc.gov/OPA

No. 01-073 June 21, 2001

NRC RENEWS LICENSE FOR ARKANSAS NUCLEAR ONE, UNIT 1 FOR AN ADDITIONAL 20 YEARS

The Nuclear Regulatory Commission has renewed the operating license for Unit 1 of the Arkansas Nuclear One nuclear power plant near Russellville. It is operated by Entergy Operations, Inc.

The Commission unanimously approved the license extension following a review of staff recommendations.

Entergy submitted an application to the NRC on January 31, 2000, to renew the license for Arkansas Nuclear One, Unit 1, which expires on May 20, 2014. The NRC conducted an extensive review of the license renewal application in accordance with Parts 51 and 54 of Title 10 of the Code of Federal Regulations.

The NRC's environmental review, under Part 51, is described in a site-specific supplement to the NRC's "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," (NUREG-1437, Supplement 3). In this Final Environmental Impact Statement, issued on April 12, the staff concluded that there were no impacts that would preclude renewal of the license for environmental reasons.

In the "Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1," (NUREG-1743) issued in June, the staff concluded that there were no safety concerns that would preclude license renewal, because the licensee had demonstrated the capability to manage the effects of plant aging.

In addition, the NRC conducted two inspections of the plant to verify information submitted by the licensee.

On May 16, the Advisory Committee on Reactor Safeguards -- an independent body of technical experts which advises the Commission -- issued its recommendation that the operating license for Arkansas Nuclear One, Unit 1, be renewed. That recommendation is contained in the "Report on the Safety Aspects of the License Renewal Application for Arkansas Nuclear One, Unit 1."

Copies of these documents and others relating to the license renewal will be available at: http://www.nrc.gov/OPA/reports/renewal.htm on the agency's web site. A copy of the staff's recommendation on the renewal of Arkansas Nuclear One, Unit 1, which contains the license conditions for the facility, will be available at the same web site as well as in the NRC Public Document Room at the agency's One White Flint North Building, 11555 Rockville Pike, Rockville, Maryland; telephone

1-800-397-4209 or (301) 415-4737.

NRC renewed the operating licenses for both units of the Calvert Cliffs Nuclear Power Plant near Lusby Maryland, for an additional 20 years on March 23, 2000, and renewed the operating licenses for the three units of the Oconee Nuclear Station near Seneca, South Carolina, for an additional 20 years on May 23, 2000. The agency is currently reviewing license renewal applications for Hatch Units 1 and 2, operated by the Southern Nuclear Operating Company, near Baxley, Georgia; Turkey Point Units 3 and 4, operated by Florida Power & Light Co., near Florida City; Virginia Electric & Power Co.'s Surry Units 1 and 2, near Surry, Va., and North Anna Units 1 and 2, 40 miles northwest of Richmond; and Duke Power Co.'s McGuire Units 1 and 2, near Charlotte, North Carolina, and Catawba Units 1 and 2, near Rock Hill, South Carolina.

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Office of Public Affairs Telephone: 301/415-8200

Washington, DC 20555-001 E-mail: opa@nrc.gov

Web Site: http://www.nrc.gov/OPA

No. 01-078 July 03, 2001

NRC EXTENDS COMMENT PERIOD TO SEPTEMBER 6 FOR TURKEY POINT LICENSE RENEWAL DRAFT ENVIRONMENTAL IMPACT STATEMENT

Seeking to increase stakeholder input, the Nuclear Regulatory Commission is extending the public comment period to 75 days on the draft environmental impact statement for the Turkey Point Nuclear Plant license renewal application. Earlier, the agency had announced the comment period would be 45 days. The statement is now open for public comment until September 6, and, as previously announced, will be the subject of public meetings July 17 in Homestead, Florida, near where the facility is located.

The NRC has been reviewing the application for extension of the Turkey Point operating licenses since Florida Power & Light Company, which operates the plants, filed it in September 2000. Under NRC regulations, the original operating license for a nuclear power plant is issued for up to 40 years. The license may be renewed for up to an additional 20 years if NRC requirements are met. The current operating licenses for Turkey Point will expire July 19, 2012, for the facility's Unit 3, and April 10, 2013, for Unit 4.

On Tuesday, July 17, the NRC staff will hold two meetings to obtain comments on the draft environmental statement. The meetings will be held at the Harris Field Complex, Homestead YMCA, 1034 Northeast 8th Street in Homestead, from 1:30 to 4:30 in the afternoon, and from 7 p.m. to 10 p.m., or until all interested people have an opportunity to speak. An open house is scheduled to begin one hour before the start of each meeting.

Written comments on the draft statement will also be considered by NRC staff. Comments should be submitted either by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Mail stop T-6 D 59, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, or by Internet to TurkeyPointEIS@nrc.gov. At the conclusion of the extended public comment period on September 6, the NRC staff will consider and address the comments provided and issue a final supplement to the agency's Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants, (NUREG-1437). That supplement will contain a recommendation regarding the environmental acceptability for license renewal.

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NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF PUBLIC AFFAIRS, REGION III

801 Warrenville Road, Lisle IL 60532

No. III-01-027 June 29, 2001

CONTACT: Jan Strasma (630)829-9663/e-mail: rjs2@nrc.gov
Pam Alloway-Mueller (630)829-9662/e-mail:pla@nrc.gov

NRC STAFF PROPOSES $55,000 FINE AGAINST MICHIGAN NUCLEAR PLANT FOR FAILURE TO PROVIDE COMPLETE AND ACCURATE INFORMATION

The Nuclear Regulatory Commission staff has proposed a $55,000 fine against Nuclear Management Co., operator of the Palisades Nuclear Power Plant, for failing to provide complete and accurate information when it requested a regulatory change at the plant in February of last year. The Palisades plant is located at Covert, Michigan.

The information was submitted when the plant staff requested authorization to permanently close off one of two steam lines connected to an auxiliary feedwater pump, which is part of a backup system to remove heat from the reactor if the normal feedwater system is lost.

(The Palisades plant, formerly operated by Consumers Energy Company, is now operated by Nuclear Management Company.)

On February 5 of last year, while the plant was shut down for planned maintenance, an underground steam pipe to a steam-driven pump ruptured. The pump was shut down and the leak terminated.

The ruptured pipe was replaced, but the remainder of the underground steam line could not be fully inspected to verify its integrity. The utility decided the steam line was not needed because a second steam pipe was available to provide steam to the pump.

Since the line was no longer considered necessary, the utility requested the NRC eliminate a requirement that it be tested periodically. In its request, the utility said its past safety analysis had considered the steam line as available for use in just one situation, that of an unlikely fire in one room containing electrical cables. Other means of maintaining the reactor in a safe condition were available without using the steam line in question, the utility indicated.

Based on its review of the information supplied, the NRC granted temporary authorization to not comply with the testing requirement. Later, the agency approved the closing of the steam line and permanently eliminated the associated testing requirement.

The NRC resident inspectors, in an inspection earlier this year, found a second fire scenario in which the steam line might be needed to maintain the reactor in a safe shutdown condition following a fire. The company had not included this scenario when it requested the testing requirement be removed.

The company and the NRC staff extensively evaluated this second fire scenario and concluded that the steam line in question would not be needed to maintain reactor safety. As a result, there was no change in the agency's decision to permit elimination of the steam line and the associated testing requirement.

In notifying the company of the proposed fine, NRC Regional Administrator James E. Dyer said, "The failure to provide complete and accurate information affected the NRC's ability to perform its regulatory function." Had the agency received complete information, it would "have had substantial further inquiry or considered additional compensatory actions before making a regulatory decision," he added.

The plant staff is being cited for its failure to identify and evaluate the second fire scenario when it requested the change in NRC requirements for the steam line. Dyer noted that the agency was satisfied that the failure to provide complete information was an oversight and not a deliberate act to withhold information.

Nuclear Management Company has until July 27 to pay the fine or to protest it. If the fine is protested an subsequently imposed by the NRC staff, the company may request a hearing.

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[NRC Home Page | News and Information | E-mail]

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs
Washington, DC 20555-001

Telephone: 301/415-8200
E-mail: opa@nrc.gov
Web Site: http://www.nrc.gov/OPA

No. 01-067

June 1, 2001

NRC ISSUES ANNUAL ASSESSMENTS FOR ALL NUCLEAR PLANTS

The U.S. Nuclear Regulatory Commission has issued annual assessment letters for all operating nuclear power plants and posted them to its web site.

The assessment letters sent to each licensee are available from the NRC Office of Public Affairs, at http://www.nrc.gov/OPA/ppr on the NRC web site, and through ADAMS, the Agencywide Documents Access and Management System.

All commercial nuclear power plants (with the exception of the two D.C. Cook plants, due to their extended shutdown) are now being evaluated under the revised reactor oversight process initiated on April 2, 2000. The NRC expects to make additional refinements to the program based on lessons learned from the first year of initial implementation.

The revised reactor oversight process reflects several important themes for all of NRC's activities -- an even greater focus on safety, an effort to improve objectivity and timeliness, a commitment to stakeholder involvement, and improved transparency of agency activities for both licensees and the general public.

As part of the new program, each plant will receive an assessment letter every six months: a mid-cycle review letter and the annual assessment letter. Updated information on plant performance is being posted to the NRC web site every quarter.

The NRC is in the process of aligning the inspection and assessment cycle with the calendar year. In order to transition to a calendar year, the current inspection and assessment cycle will consist of three quarters (the second, third and fourth calendar quarters of calendar year 2001). The next annual assessment letters will be issued in March 2002, and the next mid-cycle review letters in September 2002.

Public meetings at each plant are planned. Details of the meetings, which will be open to the public, will be announced as they are scheduled.

Details about plant performance can be found at:
http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html on the NRC web site.

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200

Washington, DC 20555-001 E-mail: opa@nrc.gov

Web Site: http://www.nrc.gov/OPA

No. 01-082 July 6, 2001

NRC TO BEGIN ONE-YEAR EVALUATION OF A REVISED PROGRAM TO ASSESS PHYSICAL SECURITY AT NUCLEAR POWER PLANTS

The Nuclear Regulatory Commission is beginning a one-year pilot of the Safeguards Performance Assessment (SPA) program, a process by which a power-reactor licensee tests the effectiveness of key elements of its physical security program.

The pilot program will also allow NRC to evaluate concepts being considered for proposed revisions to NRC regulations. It will also be used to determine if the SPA has merit as a possible replacement for th Operational Safeguards Response Evaluation (OSRE), the current NRC program to assess physical security at nuclear power plants.

The SPA program provides for:

- development of "target sets," equipment essential for safe operations of a reactor, as the basis for a licensee's physical security measures;
- participation by each security shift in a minimum of one annual licensee-evaluated drill to demonstrate proficiency for key security personnel;
- quarterly drills; and
- participation in an NRC-evaluated exercise every 3 years to test the licensee's ability to protect target sets from mock acts of sabotage.

On a case-by-case basis, licensees who volunteer to participate in the SPA pilot may be exempt from an OSRE. However, NRC baseline inspections under the reactor oversight program will continue to assess licensees' ability to protect themselves from malevolent acts. NRC will continue to conduct OSREs at sites not participating in the SPA pilot program until its evaluation of SPA is completed and the Commission determines whether it is an acceptable alternative to OSRE.

Under the SPA program, full-scale force-on-force exercises would be conducted at each nuclear power plant every three years, exceeding the frequency of the OSRE program, which provides for an NRC assessment of each plant's security programs every eight years. Like OSRE, the SPA program includes provisions to address deficiencies identified through drills and exercises within the licensee's corrective action program.

NRC will choose eight volunteers from those plants that have agreed to participate in the pilot program This evaluation is part of an ongoing effort by NRC to identify more efficient and effective ways to assess security at nuclear power plants.

July 5, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - SECY-01-0060 - THE SAFEGUARDS PERFORMANCE ASSESSMENT PILOT PROGRAM

The Commission has approved the conduct of a one-year pilot of the Safeguards Performance Assessment (SPA) Program. The Commission has also approved maintaining the Operational Safeguards Response Evaluation (OSRE) program at a reduced frequency of six OSREs during FY02. The staff should provide the Commission with an assessment of the SPA pilot program following its completion.

The Commission has approved exempting a SPA pilot program participant from an OSRE as discussed in SECY-01-0060. However, the Commission has disapproved two of the staff's proposed exempting criteria. Specifically, the first criterion should be revised to reflect that the site has undergone an OSRE within the last 5 years. The second criterion (recent site inspections have not identified any adverse performance issues) should be eliminated. However, if a licensee removes itself from the SPA pilot program before it has concluded the evaluated exercise, it will become eligible for the scheduling of an OSRE consistent with established OSRE frequency guidelines and scheduling practices.

The staff should develop a communication plan which clearly describes the significant elements of the SPA program, as well as the SPA program's relationship to the OSRE program and other elements of the NRC's plant security oversight process. Also, the staff should develop a press release to accompany release of the SECY-01-0060 and this staff requirements memorandum.

The Congress should be notified of the Commission's actions before the pilot is initiated. The staff should obtain Commission approval of any Congressional correspondence related to this pilot program.

cc: Chairman Meserve
Commissioner Dicus
Commissioner McGaffigan
Commissioner Merrifield
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)

June 13, 2001

MEMORANDUM TO: Richard J. Barrett, Acting Director
Future Licensing Organization
Office of Nuclear Reactor Regulation

FROM: Eric J. Benner, Regulatory Infrastructure Project Manager ***/RA/***
Future Licensing Organization
Office of Nuclear Reactor Regulation

SUBJECT: FORTHCOMING WORKSHOP ON FUTURE LICENSING ACTIVITIES

DATE & TIME: Wednesday, July 25, 2001
9:00 a.m. - 8:00 p.m
Thursday, July 26, 2001
9:00 a.m. - 1:00 p.m.

LOCATION: U.S. Nuclear Regulatory Commission
Two White Flint North
11545 Rockville Pike
Rockville, Maryland 20852
TWFN Auditorium

PURPOSE: To inform the public of the current and proposed activities of the NRC staff regarding future applications and to solicit public concerns and feedback on identified issues and challenges. See attached preliminary agenda.

PARTICIPANTS*	NRC	PUBLIC
	F. Cameron R. Barrett J. Lyons M. Gamberoni et al.	Any interested member of the public

Attachment: Preliminary agenda

CONTACT: Eric J. Benner, NRR
301-415-1171

Members of the public may pre-register for this meeting by contacting Eric Benner at (800) 368-5642, ext. 1171, or by Internet at ejb1@nrc.gov by July 20, 2001.

Preliminary Agenda

Recently, the nuclear industry has indicated that they may be submitting licensing applications in accordance with Parts 50 and 52 of the *Code of Federal Regulations* (10 CFR Parts 50 and 52) in the near future with the intent to build and operate new nuclear power plants. These submittals could include applications for Early Site Permits, Design Certifications, Combined Licenses, and Operating Licenses. Additional activities could include pre-application reviews related to these submittals and requests to reactivate Construction Permits to allow the applicant to resume construction of nuclear facilities.

The purpose of this workshop is to inform the public of the current and proposed activities of the NRC staff to prepare for these potential future licensing applications, discuss the mechanisms available to the public for providing input during these licensing activities, and to solicit public feedback on identified issues and challenges.

A final agenda and schedule will be published on the NRC future licensing web site when it is available: **http://www.nrc.gov/NRC/REACTOR/FLO/index.htm**

Workshop Organization:

For each of the agenda topics, the NRC staff will describe the activity and the regulatory process governing the activity, discuss the actions the NRC is taking to prepare for the anticipated submittal, and provide an estimate of expected milestone dates, where available. The staff will discuss how the public can provide input to the staff during different stages of the activity. The remainder of the allotted time will then be used for an open dialogue among all workshop participants. The NRC staff will then briefly summarize the identified concerns.

In addition, the workshop will contain two open sessions to allow public participants to discuss issues not addressed or to have further discussions on the agenda items. The first of these sessions will be on the evening of July 25 to allow participation by individuals who cannot attend during the work day and will include a brief presentation by the NRC staff summarizing all of the agenda items. The second of these sessions will be held on the morning of July 26.

Following the workshop, the NRC staff will summarize and document the identified concerns. The summary will be sent to those workshop participants who request a written copy, as well as the Commission, and will be posted on NRC's future licensing website.

Agenda Topics:

Licensing Processes

Early Site Permits

10 CFR Part 52 allows an applicant to apply for an early site permit, which provides for resolution of site safety, environmental protection, and emergency preparedness issues, independent of a specific nuclear plant review. The early site permit application must address the safety and environmental characteristics of the site and evaluate potential physical impediments to develop an emergency evacuation plan.

Design Certification

The NRC may certify a standard plant design through a rulemaking under 10 CFR Part 52, independent of a specific site. Among other requirements, the design certification applicant must demonstrate that its design complies with current NRC regulations, provide a probabilistic risk assessment, and provide a proposed set of inspections, tests, analyses, and acceptance criteria that will demonstrate that the plant will operate in accordance with the design certification.

Combined License

A combined license, issued under 10 CFR Part 52, authorizes construction of the facility and specifies the inspections, tests, and analyses that the licensee must perform. It will also specify the acceptance criteria that, if met, are necessary to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license and the applicable regulations.

Construction Inspection

The NRC is reactivating the construction inspection program revision effort suspended in 1994. This effort will include review and revisions of applicable inspection guidance and training for inspection of critical attributes of construction processes and activities.

Reactivation of Construction Permits

A licensee has indicated that it may consider resumption of construction and application for operation of an unfinished nuclear power plant. Resumption of construction would require the normal NRC reviews and inspections under the existing Construction Permit and application for operation would require the normal NRC reviews for an Operating License (OL) for the facility.

Regulatory Infrastructure

The NRC staff is planning to update some of its regulations to facilitate its review of new applications. The Nuclear Energy Institute is considering proposing a New Plant Regulatory Framework.

Current Activities

Readiness Assessment, Organizational Development, and Staffing

The NRC is performing a readiness assessment to develop postulated licensing scenarios for future application reviews, including resource estimates and critical skills. In addition, the NRC is determining the appropriate organizational structure for future licensing activities and will be staffed accordingly.

Rulemaking

The NRC is currently revising several rules: (1) a revision to 10 CFR Part 52 to clarify and incorporate lessons learned from previous design certifications; (2) a revision to 10 CFR Part 51 to address higher enrichment fuel; and (3) a revision to 10 CFR Part 51 to clarify the scope of alternative site reviews when applying for an Early Site Permit.

Pre-application Reviews

The NRC Policy Statement on advanced reactors encourages early interaction between applicants and vendors with the NRC. To that end, the following four applicants and vendors have initiated interactions with the NRC regarding different advanced reactor designs:
(1) Westinghouse "AP1000," (2) Exelon "Pebble Bed Modular Reactor" (PBMR),
(3) Westinghouse "International Reactor Innovative and Secure" (IRIS), and (4) General Atomics "Gas Turbine-Modular Helium Reactor" (GT-MHR).

RISK-INFORMING 10 CFR 50.46

Presented to
Advisory Committee on Reactor Safeguards
(Full-committee)

Presented by
Mary Drouin and Alan Kuritzky
RES/DRAA/PRAB
U.S. Nuclear Regulatory Commission
(301) 415-6189

July 11, 2001

OUTLINE

- Purpose/goal of meeting
- Background - Option 3
- Tentative Recommendations and schedule
- Activities
  - Feasibility assessment of changing 10 CFR 50.46
  - Feasibility assessment of additional changes to 10 CFR 50.46
  - Other Option 3 activities
- Status and schedule

PURPOSE/GOAL OF MEETING

- Provide status report on staff's efforts to risk-inform 10 CFR 50.46 (Paper currently pre-decisional)
- Solicit feedback and comments from ACRS:
  - Options
  - Implementation issues
  - Feasibility
- Letter requested

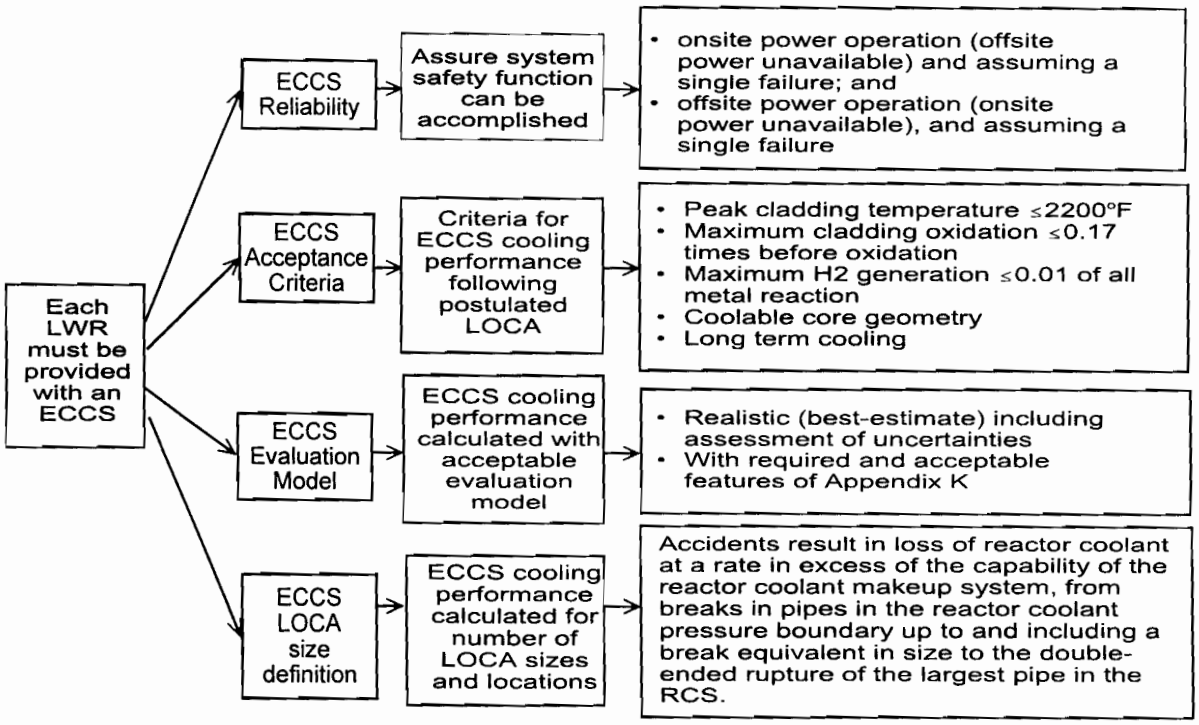
BACKGROUND

SECY-99-264 (Nov 9, 1999) defined plan for Option 3 work

OPTION 3 FRAMEWORK:

- Phase I:
  - Part A: Identify candidate requirement
  - Part B: Prioritize
  - Part C: Evaluate feasibility and provide recommendations to Commission
    - ★ Develop technical content and basis for alternative
    - ★ Identify policy issues
    - ★ Identify required technical work
    - ★ Identify required resources
- Phase II:
  - Part A: Perform technical work
  - Part B: Develop and implement rulemaking

OVERVIEW OF 50.46 (including Appendix K and GDC 35)



FEASIBILITY ASSESSMENT OF CHANGING 10 CFR 50.46 (including Appendix K and GDC 35)

- Changes to reliability, acceptance criteria and evaluation model feasible
  - ECCS reliability resulting from technical requirements not commensurate with risk significance of the various LOCA sizes
  - Unnecessary conservatisms exist in the requirements
- Changes to spectrum of LOCA sizes definition more complex
  - Current estimates of the frequency of large-break LOCAs are uncertain and are not low enough to allow elimination of all large-break LOCA sizes from the design bases

FEASIBILITY ASSESSMENT OF CHANGING 10 CFR 50.46 (cont'd)

- Short-term considerations:
  - A. Changes to the technical requirements of the ***current*** 50.46 related to acceptance criteria and evaluation model
  - B. Development of a voluntary risk-informed ***alternative*** to the reliability requirements in 50.46
- Long-term considerations:
  - Evaluation of the definition of the spectrum of break sizes
- Follows the guidelines in Option 3 framework
- Framework is designed to ensure that changes are risk-informed, and include consideration of defense-in-depth principles

SCHEDULE

A. Modification of the existing 10 CFR 50.46 and Appendix K:
- Develop proposed rule — 12 months from date of SRM or 2 months after completion of technical work (whichever is later)
- Perform technical work — On or before July 2002

B. Development of a risk-informed alternative to 10 CFR 50.46, Appendix K and GDC 35:
- Develop proposed rule — 12 months from date of SRM or 2 months after completion of technical work (whichever is later)
- Perform technical work — On or before April 2002

- Continue longer-term feasibility assessment on additional changes to 50.46, including rigorous analysis of LOCA frequencies
  - Up to 3 years

A. POSSIBLE CHANGES TO THE CURRENT 50.46

- Replace the current prescriptive ECCS acceptance criteria in 50.46 with a performance-based requirement

- This requirement would:
  - demonstrate adequate post- quench cladding ductility and adequate core-coolant flow area to ensure that the core remains amenable to cooling, and,
  - for the duration of the accident, maintain the calculated core temperature at an acceptably low value and remove decay heat.

- Allows use of cladding materials other than zircaloy or ZIRLO without licensees having to submit an exemption request

A. POSSIBLE CHANGES TO THE CURRENT 50.46 (cont'd)

- Revise the requirements for the ECCS evaluation model to be based on more realistic analyses
- Specifically this update could involve:
  - replacing the current 1971 American Nuclear Society (ANS) decay heat curve with a model based on the 1994 ANS standard.
  - replacing the current decay heat multiplier of 1.2 with an NRC-prescribed uncertainty treatment.
  - deleting the limitation on PWR reflood steam cooling for small reflood rates.
  - replacing the Baker-Just zirconium steam model with the Cathcart-Pawel zirconium steam oxidation model for heat generation.
  - deleting the prohibition on return to nucleate boiling during blowdown.
- Rule requirements would include a provision that would account for recognized nonconservatisms and model limitations

A. POSSIBLE CHANGES TO THE CURRENT 50.46 (cont'd)

Additional technical work would be required to support the actual rule changes

- Support removal of unnecessary conservatisms from Appendix K
- Develop guidelines for demonstrating adequate post-quench ductility as a replacement for the current prescriptive acceptance criteria
- Support development of the regulatory guides needed for implementing the modifications to the existing rule

B. DEVELOP A VOLUNTARY RISK-INFORMED ALTERNATIVE 50.466

- Include technical requirements to ensure an ECCS reliability that is commensurate with the frequency of challenge to systems

- Two options to accomplish ECCS system reliability (in place of the simultaneous loss of offsite power requirement and single failure criterion):

  1. A deterministic system reliability requirement based on risk information
     - e.g., an ECCS design requirement that only one train of ECCS is required for LOCAs larger than a specified size

  2. An ECCS functional reliability requirement that is commensurate with the LOCA frequency
     - e.g., a requirement that ECCS design must be such that the core damage frequency [CDF] associated with a specified set of LOCAs is less than an NRC-specified CDF threshold

B. DEVELOP A VOLUNTARY RISK-INFORMED ALTERNATIVE 50.46 (cont'd)

Additional technical work would be required to support the actual rule changes

- Determine acceptable methods and assumptions for performing LOCA CDF and ECCS reliability analyses for those alternatives requiring such analyses

- Determine appropriate reliability and CDF threshold values

- Identify features that tend to decrease the likelihood of loss of offsite power following a LOCA

- Determine acceptable methods and assumptions for estimating plant-specific probability of loss of offsite power given a LOCA.

- Support development of the regulatory guides needed for implementing the recommended risk-informed alternative rule

POSSIBLE LONGER-TERM FEASIBILITY ASSESSMENT OF ADDITIONAL CHANGES TO 50.46

- Additional changes to 50.46 may also have merit:
  - evaluation of the definition of the spectrum of breaks and locations
- The extent of potential change to the definition of pipe break size is dependent on the state-of-knowledge of the frequency of LOCAs of various break sizes
- For example, if a set of LOCAs can be demonstrated to have a collective mean frequency of occurrence of below —
  - 10^{-4}/yr, some regulatory relief may be appropriate
  - 10^{-5}/yr, may be appropriate to remove these LOCAs from the plant design basis, with some mitigative capability
  - 10^{-6}/yr, may be appropriate to remove these LOCAs from the plant design basis
- Staff to continue to perform the technical work to determine its feasibility

POSSIBLE LONGER-TERM FEASIBILITY ASSESSMENT OF ADDITIONAL CHANGES TO 50.46 (cont'd)

- The staff will continue to meet with representatives of the nuclear industry in public meetings to address and resolve the technical issues
- These issues include, for example,
  - initial flaw distributions, degradation mechanisms, material response and uncertainty analysis
- If found feasible, the staff would recommend additional changes, potentially including rulemaking to change the wording in 50.46 and Appendices A and K of Part 50 which would allow the licensee to use an alternate pipe size, subject to some level of NRC approval

OTHER OPTION 3 ACTIVITIES

- GDC 35 requires that the ECCS safety function be accomplished assuming a single failure
- Considering replacing this single failure criterion in the alternative rule, but only as it affects ECCS
- The single failure criterion is applied to more than just the ECCS. GDCs 17, 34, 38, 41 and 44 also contain the single failure criterion.
- A generic change to the Part 50 Appendix A single failure criterion definition may be warranted
  - Staff intends to assess the feasibility of a single generic change under Option 3
  - Such a risk-informed definition would also address the Commission's guidance in the SRM of February 3, 2000
- The staff has also begun to investigate changes to the special treatment technical requirements of Part 50
- The staff has deferred further work on this to better focus its resources on assessments of 50.44 and 50.46, but would reassess its priority late this year

STATUS AND SCHEDULE

- Paper pre-decisional
- Requesting letter from ACRS
- Short-term change:
  - Develop proposed rule — 12 months from SRM or 2 months after technical work (whichever later)
  - Perform technical work
    - Modify current 50.46 — On or before July 2002
    - Alternative 50.46 — On or before April 2002
- Longer-term Option:
  - Up to 3 years

(4)

Industry Views on SFP Risk Study and Policy Options

ACRS

July 11, 2001

NEI

Overview

- Impacts of not completing risk study
- What's needed to complete the study (Dr. Henry, Fauske and Associates)
- Industry Views on Policy Options
- Recommendations

NEI

Risk Study Should be Completed

- Failure to complete has impacts on:
  - Policy options for decommissioning rules
  - Use of results in plant PRAs (<3E-6??)
  - Accuracy of value/impact and backfit analyses
  - Unrealistic conclusions (e.g., cask drop results) being applied elsewhere (PRA for casks)

NEI

Safeguards Option

- Protect against the DBT to a performance standard of 5 rem to the public or,
- Demonstrate through plant specific analyses that a zirconium fire is <u>precluded</u>. Includes:
  - Design features, heat-up analysis, or mitigating actions including response by law enforcement before fire commences

NEI

Issues/Safeguards Option

- No opportunity to comment on guidance for performing plant specific analyses
- Any change to adversary characteristics could invalidate the entire program
- Standard to "preclude" a zirconium fire not reflected in EP policy analyses

NEI

Insurance Option

- 60-days after shutdown:
  - Primary coverage reduced to $100M
  - Exempted from participation in secondary pool
  - On-site property damage not required
- Facility must comply with staff assumptions and industry commitments

NEI

Issues/Insurance Option

None, appears to be rational, risk informed approach.

NEI

EP Option

- Some reduction in offsite EP in the first year
- Elimination of offsite EP at 5 years.
- Licensee must comply with staff assumptions and industry commitments

NEI

Issues/EP Option

- Can't quantify benefits without more information on reduction in offsite requirements
- Defense in depth?? May be non existent if adhoc EP is just as effective shortly after shutdown and/or if evacuation is ineffective for large seismic events
- Cask drop is not a realistic event
- Sabotage already uses standard of "precluding" the event

NEI

Recommendations

- Complete the risk study, peer review results and derive a best estimate using existing data on cask drops, Ruthenium release and EPRI seismic numbers
- Revisit conservatism in seismic checklist
- Ensure EP reductions are commensurate with the risk and quantifiable defense in depth

NEI

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT:	**SECY-01-100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools"**
DATE:	**July 11, 2001**
PRESENTER:	**Bill Huffman**
PRESENTER'S TITLE:	**Project Manager, Division of Regulatory Improvement Programs Office of Nuclear Reactor Regulation**
PRESENTER'S TEL. NO.:	**(301) 415-1141**

Policy SECY Background

- Decommissioning nuclear power plants seek early regulatory relief from various Part 50 requirements. Three areas involve consideration of zirconium fire:
  - Insurance
  - Security
  - Emergency Preparedness (EP)

- Relief provided by exemption process

- Several rulemaking attempts initiated
  - Stopped; technical bases inadequate

- Industry challenged zirconium fire criteria

- Commission meeting March 17, 1999
  - SRM sanctioned risk-informed approach

- Staff committed to perform detailed technical study on decommissioning plant spent fuel pool accident risk

- Risk study now complete (NUREG-1738)

RISK STUDY FINDINGS

- Spent fuel pool accident risk is low
- Agency quantitative health objectives are met; risk well within Commission's safety goals
- Risk findings can be used consistent with RG 1.174 guidelines for small increase in risk
  - e.g., small change in risk if offsite EP is relaxed at decommissioning plants
- Cannot define a generic decay heat time beyond which a zirconium fire is not physically possible

Policy Issue 1

Policy Issue 1 Should the Safety Goals for Operating Nuclear Power Plants be Applied to Decommissioning Plants?

Options: (1) Yes

(2) No

Recommendation: Option (1)

Apply the Commission safety goal policy statement to decommissioning nuclear power plants storing spent fuel in the spent fuel pool (SFP)

- Operating plant safety goals are appropriate since consequence from a postulated SFP zirconium fire can be similar to a large early release event at an operating reactor

- Permits application of SFP risk study, NUREG-1738, and existing risk-informed decision making guidance to decommissioning plant regulatory improvements

Policy Issue 2

Policy Issue 2 Should the Commission develop an approach using probabilistic risk assessments for quantifying the likelihood of sabotage?

Options: (1) Commit resources to begin development of a PRA methodology that can be used to assess the likelihood of sabotage

(2) Evaluate current state-of-art PRA methodologies for assessing sabotage and determine if further development is warranted

(3) Continue to assess likelihood of sabotage in a qualitative manner using deterministic and performance-based safeguards design criteria

Recommendation: Option (3)

The approach for developing new safeguards regulatory requirements for decommissioning plants will be based on deterministic and performance-based criteria because:

- Methods for estimating the likelihood of sabotage is considered to be beyond the state of the art of PRA

- Attempting to develop PRA methods for estimating the likelihood of sabotage would require substantial resources and cannot be done unilaterally by the NRC

Policy Issue 3

Policy Issue 3 What safeguards protection goal should the Commission apply to SFPs at decommissioning plants?

Options: (1) Maintain a level of security commensurate with that of an operating plant

- Design criteria - protect against radiological sabotage by the design basis threat (DBT)

(2) Apply a performance-based protection goal for spent fuel stored in decommissioning SFPs as recommended in the newly proposed rule changes for physical protection at nuclear power reactors in SECY-01-101, dated June 4, 2001

- Design criteria - protect against radiological sabotage by the design basis threat (DBT)

- Performance standard - no fuel damage that exceeds the offsite dose limits of 10 CFR 72.106

Policy Issue 3 (cont.)

(3) Apply the protection goal for an independent spent fuel storage installation (ISFSI)

- Design criteria - protect against radiological sabotage that results in a loss of control of the facility

- Performance standard - no fuel damage that exceeds the offsite dose limits of 10 CFR 72.106

Recommendation: Option (2)

Applies the performance-based protection goal recommended in the proposed revision to 10 CFR 73.55 in SECY-01-101

- Appropriate level of physical protection for decommissioning plant SFPs (provides transition between operating reactors and dry cask ISFSIs)

- Should provide sufficient flexibility to permit a decommissioning licensee to focus their security program and response strategies in a manner that reduces regulatory burden below operating reactor levels (Option 1) while maintaining safety

Policy Issue 4

Policy Issue 4 What level of insurance is appropriate for licensees of decommissioning plants with fuel stored in the SFP?

Options:

(1) Maintain insurance at operating reactor levels until all spent fuel is removed from the SFP

(2) Maintain insurance at operating reactor levels until a plant-specific thermal-hydraulic heatup analysis demonstrates that uncovered spent fuel would not reach zirconium ignition temperature

(3) Relax insurance after a generic fixed period of time based on qualitative policy judgment that zirconium fires are unlikely based on decay time alone (although still possible)

(4) Relax insurance requirements shortly after permanent shutdown based on the low generic frequency of events leading to a zirconium fire contingent on implementation of certain SFP design, operational, and administrative features committed to by the industry or assumed by the staff in the risk study (these controls are referred to as industry decommissioning commitments - IDCs and staff decommissioning assumptions - SDAs in the policy SECY)

Policy Issue 4 (cont.)

Recommendation: Option (4)

Since the presence or absence of insurance has no effect on the probability or consequences of a zirconium fire, reducing insurance does not increase the radiological risk to the public. Reducing insurance coverage shortly after permanent shutdown is justified based on the low likelihood of events leading to a zirconium fire (contingent on implementation of the risk study IDCs and SDAs)

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT:	SECY-01-100 Emergency Preparedness Policy Issue
DATE:	July 11, 2001
PRESENTER:	R. L. Sullivan, CHP
PRESENTER'S TITLE:	EP Specialist Division of Inspection Program Management Office of Nuclear Reactor Regulation
PRESENTER'S TEL. NO.:	(301) 415-1123

Policy Issue 5

Policy Issue 5 What level of offsite emergency preparedness (EP) is appropriate for decommissioning plants given the low likelihood of a radiological release large enough to exceed protective action guides offsite?

Options:

(1) Substantially reduce or eliminate offsite EP requirements shortly after permanent shutdown based on the low generic frequency of events leading to a zirconium fire (contingent on implementation of IDCs and SDAs)

(2) Maintain offsite EP at operating reactor levels until all spent fuel is removed from the SFP

(3) Modify the level of offsite EP required at decommissioning plants based on sufficient time to take ad hoc mitigative and protective actions before a large release can begin

Recommendation: Option (3)

Incrementally reduce and eventually eliminate offsite EP for decommissioning plants based primarily on sufficient time to implement ad hoc protective and mitigative actions.

Maintains Commission defense-in-depth philosophy and is risk-informed by a reasonable assurance that the likelihood of a zirconium fire event is very low (contingent on implementation of the risk study IDCs and SDAs)

Incrementally Reduce EP

- Maintain full scope EP for a period not expected to exceed one year

- Eliminate portions of EP requirements IAW the physics of the spent fuel e.g., little Iodine, no rapidly evolving accidents, no need for rapid multi-discipline engineering assessment

Based on:

- The length of time available for protective actions before a zirconium fire can begin,
- the length of time available for and relative simplicity of mitigative actions,
- the effectiveness of protective measures implemented by trained public agencies, and
- the very low frequency of initiating events that can cause a zirconium fire when IDCs and SDAs are implemented.

Eventually Eliminate EP

- When fuel is decayed such that the fuel can not reach ignition temperature for at least 10 hours (but not more than 24 hours)

- Require offsite EP at a level similar to that in 10 CFR 72.32 for MRS facilities

Based on:

- The effectiveness of ad hoc protective measures for the protection of the public, especially when there is time to prepare

(5)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
484	4	4-1

Title **Need to Revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants**

Author: **Mario V. Bonaca**

Letter dated June 26, 2001, from David Lochbaum, Union of Concerned Scientists, to Christopher Grimes, NRR, SUBJECT: Revision to the License Renewal Rule

4

Instructions to Preparer
1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person
N. Dudley

Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

June 26, 2001

Mr. Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555-0001
Earth

Dear Mr. Grimes:

Thank you for arranging for the Union of Concerned Scientists to have a seat at the table during the license renewal meeting scheduled for Thursday, June 28, 2001. Unfortunately, I will not be able to attend this meeting due to a matter that just recently developed. I had prepared three topics that I planned to cover during the meeting. Those topics are:

1. Gaseous and liquid radwaste systems: In May 2000, UCS submitted a petition for rulemaking seeking to revise the scope of the license renewal rule to cover those portions of the gaseous and liquid radwaste systems whose failure could potentially cause excessive releases of radioactivity to the environment. The justification that accompanied our petition provided some examples of credible equipment failures. We continue to believe that the rule needs to explicitly address these vulnerabilities.

2. Adequacy of aging management programs: During the session on license renewal that you chaired at the 2001 Regulatory Information Conference, I presented data on eight unplanned reactor shutdowns since January 1, 2000, due to equipment failures caused by aging. That list has been amended by an additional shutdown. Given that the primary purpose of aging management programs is to monitor the condition of important equipment and structures so as to effect repairs and replacements before failures occur, these reactor shutdowns indicate that the programs may not be achieving the expectations. We think that the data suggest that the license renewal rule, or the associated regulatory guidance, needs to be made more explicit with respect to the criteria defining acceptable minimum standards for aging management programs.

3. One-time inspections: At the workshop last fall and the subsequent Commission briefing, UCS conveyed a concern about one-time inspections. Today, the NRC grants license renewal applications predicated on the assumption that the one-time inspections will confirm negligible degradation. But what if these 'confirmatory' inspections reveal problems when the inspections are finally conducted years later? The licenses will have already been renewed and the plant owners may cry "Backfit!" when the NRC requests reasonable efforts based on

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2397 Shattuck Avenue · Suite 203 · Berkeley, CA 94704-1567 · TEL: 510.843.1872 · FAX: 510.843.3785

the newly acquired knowledge.[1] The license renewal rule, or its associated regulatory guidance, may need to be made more explicit with respect to the staff's authority in dealing with one-time inspection surprises.

Sincerely,

David Lochbaum
Nuclear Safety Engineer
Washington Office

[1] A representative of an industry group has already presented to the Commission his belief that the NRC must go through the backfit rigmarole before asking any plant owner follow voluntary initiatives.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

July 11, 2001

LICENSE RENEWAL RULEMAKING RECOMMENDATIONS

Sam Lee, NRR/DRIP/RLSB

(6)

COMMISSION REQUEST

August 27, 1999, Staff Requirements Memorandum (SRM) responding to SECY 99-148, "Credit for Existing Programs for License Renewal":

> **"[T]he staff should prepare a detailed analysis and provide recommendations to the Commission on whether it would be appropriate to resolve generic technical issues, including any credit for existing programs, by rulemaking."**

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) COMMENT (April 13, 2001, ACRS letter)

ACRS Comment	Staff Recommendation
Include results of scoping process in license renewal applications	Clarify guidance documents

UNION OF CONCERNED SCIENTISTS (UCS) COMMENTS
(June 26, 2001, UCS letter)

UCS Comment	Staff Recommendation
Radwaste systems should be covered in scope of license renewal	Address under rulemaking petition process
Define criteria for acceptable minimum standards for effective aging management programs	Clarify guidance documents
Define basis for reliance on one-time inspections	Clarify guidance documents

NUCLEAR ENERGY INSTITUTE (NEI) COMMENT
(June 4, 2001, NEI letter)

NEI Comment	Staff Recommendation
Rulemaking is not necessary at this time	Agree

STAFF RECOMMENDATIONS

- Rulemaking is not necessary at this time
- Clarify renewal guidance to address comments
- Continue to monitor renewal lessons and other rulemaking for opportunities to improve process

NRC PROPOSED BULLETIN TO ADDRESS:

CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

Jack Strosnider

US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Engineering

Meeting with
Advisory Committee on Reactor Safeguards

July 11, 2001

6

SAFETY PERSPECTIVE

- Failure of a CRDM nozzle constitutes a LOCA and control rod ejection (REA)
- Existing PRAs indicate a level of risk requiring increased attention
- Worst case crack found at a high susceptibility plant had a remaining ligament factor of safety of ≈6 to failure
- No reason to conclude that cracking won't affect additional units
- Timely, effective inspections should provide additional information on extent of the problem and provide confidence that safety is maintained and regulatory requirements are satisfied
- CRDM nozzle failure not expected to challenge containment integrity

TECHNICAL ISSUES HIGHLIGHTED BY ACRS SUBCOMMITTEES

TECHNICAL ISSUE	BULLETIN APPROACH
Susceptibility Model Uncertainties	Uses rankings as basis for graded approach regarding appropriate inspection method qualification and level of information requested -- information should provide greater insights and support assessment of need for additional regulatory actions
Effectiveness of Visual Inspections	Provides qualification criteria for plant-specific evaluation (availability of deposits on head, discrimination of VHP nozzle deposits, etc.), in a graded approach appropriate to relative susceptibility ranking
Evaluation of Crack Growth Rate and Annulus Chemistry	Licensees will need to provide basis for annulus chemistry and crack growth rate if they rely on analysis for basis of no inspection or lesser inspection

INDUSTRY AND BULLETIN APPROACHES TO INSPECTION

ITEM	BULLETIN APPROACH	INDUSTRY APPROACH
Examination Method	Graded approach**: (1) volumetric for plants that have leaked, (2) plant-specific visual qualification for high susceptibility plants (< 4 EFPY from Oconee 3), (3) VT-2 visual qualification for moderate susceptibility plants (from 4 to 30 EFPY of Oconee 3)	Visual examination ("capable of detecting small amounts of boric acid deposits") of plants < 10 EFPY of Oconee 3; continue boric acid walkdowns for other plants
Plants Affected	(1) 4, (2) 10, (3) 31 = 45	25
Timing	High susceptibility plants by end of 2001** (6 of 14 high susceptibility plants do not have outages scheduled before 12/31/01)	Next RFO
Sample Size	100% of VHP nozzles	100% visual of VHP nozzles
Expansion Criteria	On detection of leakage, volumetric examination of 100%**	Not specified (ASME Code criteria - 1:1)

** Or alternative approach justified by the licensee

RISK ASSESSMENT

- LOCA/reactivity insertion
  - LOCA - mitigating strategy is well understood
    - break location means operator can more readily manage coolant inventory (longer time to switch to recirculation)
  - REA - single rod ejection (hot zero power); core damage is unlikely
    - multiple rod ejection needs to be assessed

- Collateral issues
  - Need to assess the effect of multiple rod ejection accident
  - LOCA with multiple rod fail to insert
  - Recirculation-related issues

- Probabilistic fracture mechanics (PFM)

- Containment integrity is not challenged

ADDITIONAL WORK

- Complete work of RES expert group
- NRR user need request to RES (June 5, 2001)
  - NDE/ISI
  - Crack growth in Inconel weld metal (INCO 82/182)
  - Crack growth in Inconel base metal (Alloy 600) nozzles, considering chemistry of annulus
  - Residual stresses
  - Viability of visual leakage detection from CRDM nozzles and weld PWSCC cracks
  - Repairs and mitigation
  - Susceptibility models (base and weld metal)
- Risk insights and additional sequence delineation
- Continued review of industry activities

NRC PERFORMANCE GOALS

- Maintain Safety
- Reduce Unnecessary Burden
- Improve Regulatory Efficiency and Effectiveness
- Increase Public Confidence

7

MRP - Alloy 600 ITG RPV Penetrations

Presentation To ACRS Subcommittees
July 10, 2001

Purpose

- Industry Goals:
  - Near Term: Assure Structural Integrity
  - Longer Term: Develop Program to Manage PWSCC
- Explain Background of Head Penetration Issue
- Present Status of MRP Program
- MRP Recommendations for Industry

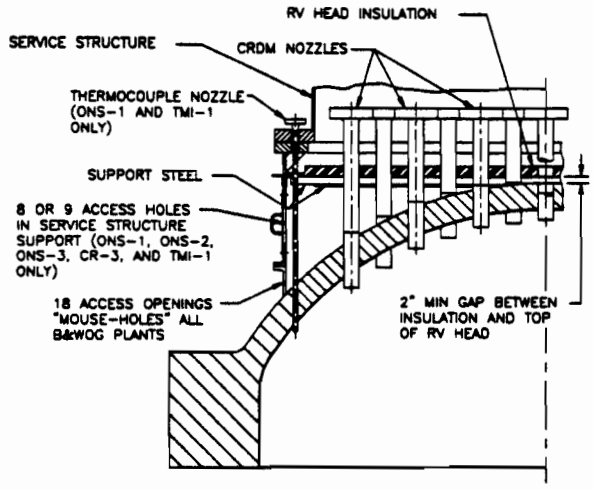
RPV Penetration Summary

- Near Term Conclusions:
  - Axial PWSCC in CRDM nozzles does not impact plant safety
    - Bounded by previously submitted Safety Assessments (1993/94)
  - Reasonable assurance that other PWRs do not have circumferential cracking that would exceed structural margin
    - Oconee and ANO-1 in highest grouping based on effective time-at-temperature
    - Leaks discovered by careful visual inspection of top head surface
    - Volumetric examination of other nozzles found only minor craze cracks
    - Leaks discovered with significant structural margin remaining
    - Several other plants in highest groupings have no evidence of leakage

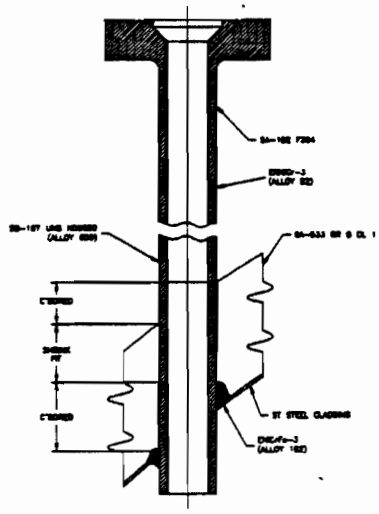
Other Ongoing MRP Activities

- Risk Assessments
- Probabilistic Fracture Mechanics
- Assessment of Crack Growth Data and Needs
- NDE Demonstration
  - Block Design and Fabrication
  - Technique Development and Demonstration
- Information and Training Package for Visual Examination
- Flaw Evaluation Guidelines
- Review of Repair and Mitigation Strategies

Side View Schematic of B&W-Design Reactor Vessel Head, CRDM Nozzles, Thermocouple Nozzles, and Insulation



Schematic View of B&W-Design CRDM Nozzle Area



Issue Background

- Bugey-3 cracking in 1991 characterized as:
  - ID-initiated, through-wall axial flaws
  - Through-wall axial flaw initiated OD circumferential flaw in RV head penetration crevice
- Lack of fusion detected in attachment welds at Ringhals-2 (1992)
- Industry safety assessments prepared (early 90's) for these types of cracking
- Additional European PWRs Discovered Axial Penetration Cracks and Initiated Head Replacements
- DC Cook 2 Found and Repaired a Single Cracked Penetration (1994)
- Owners Groups Programs to Manage for Their Units

Background: GL 97-01

- GL 97-01 Issued April 1, 1997
- Owners Groups Prepared Generic Responses
- Responses Coordinated Between Owners Groups by NEI Task Force
  - Histogram Ranked Plants, Normalizing Both Industry Models to DC Cook 2
- Individual Utilities Supplied Information for Their Plants
- Lead Plants Scheduled for Inspections Based on Histogram
  - ET for Detection
  - UT for Sizing of ID Flaws

Background: GL 97-01 Histogram

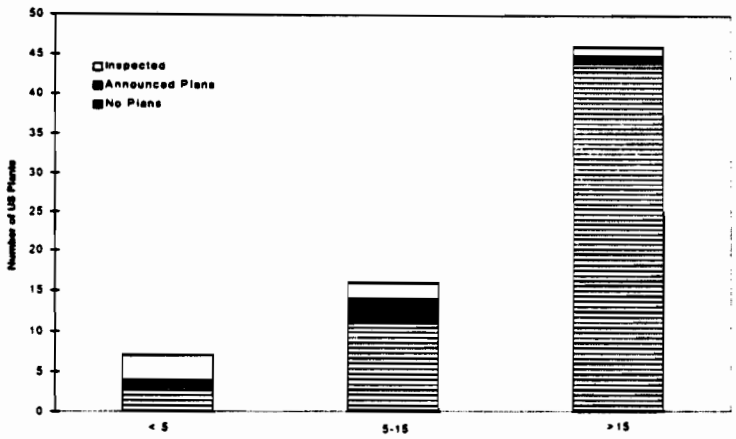
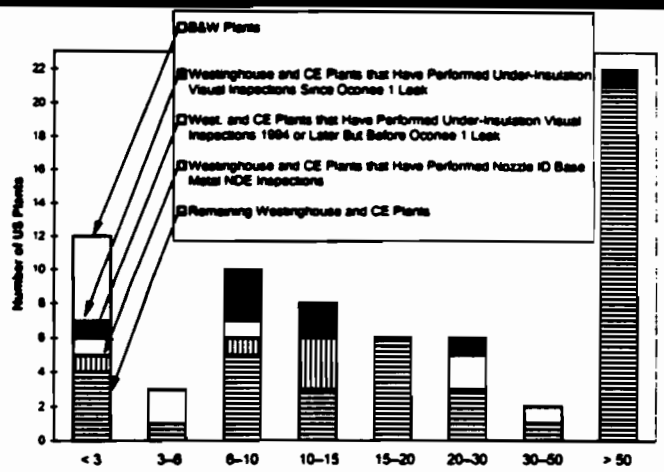


Figure 1: Effective Full Power Years, Measured from January 1997

Recent Experience

- Recent **J-groove Weld and OD-initiated** Cracking Observed at B&W-Design Plants
  - ONS-1 (November 2000)
  - ONS-3 (February 2001)
  - ANO-1 (March 2001)
  - ONS-2 (April 2001)

Time-Temperature Histogram Chart in MRP-44 Part 2 Interim Safety Assessment



Recent Experience

Oconee Experience

- Visual inspection of Unit 1 RV head identified small amounts of boron accumulation at the base of CRDM nozzle 21 and several T/C nozzles.
- Visual inspection of Unit 3 reactor vessel head identified small amounts of boron accumulation at the base of several CRDM nozzles. The suspect nozzles were #'s 3, 7, 11, 23, 28, 34, 50, 56, 63.
- Visual inspection of Unit 2 reactor vessel head identified boron accumulation at the base of CRDM nozzle #'s 4,6,18, and 30

Oconee Background Information

- Modifications to cut access ports (9 each - 12 in diameter) into the Oconee service structure were completed during outages in Spring 1994, Spring 1993, and Fall 1994 for Units 1, 2, and 3 respectively.
- Modifications to service structure allowed access to domed portion of head for bare metal inspections and wash down of the head to remove old boron deposits.

Oconee
Background Information

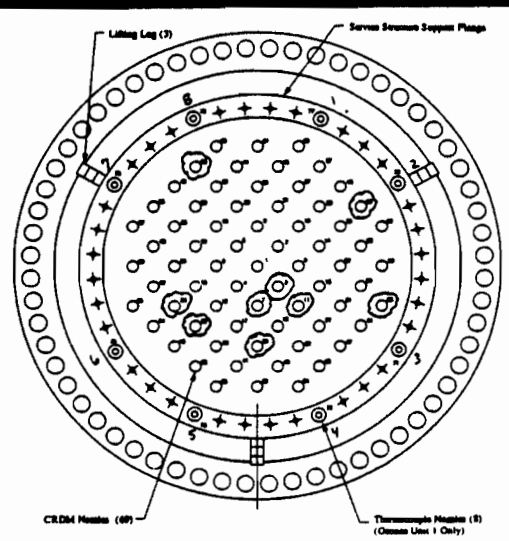
- T/C nozzles installed in Unit 1(only) for instrumentation purposes, but were never put into service.
- Located outboard of the CRDMs and fabricated from 0.75" Schedule 160 Alloy 600 pipe
- Material Specification is SB-167 and procured from Huntington Alloys as cold drawn, ground, and annealed pipe
- Procured to 1965 ASME B&PV Code

ONS-1 RV Head Showing Boric Acid At Thermocouple Nozzle

Oconee
Background Information

- CRDM (69) nozzles are constructed of Alloy 600 and procured in accordance with requirements of SB-167, Section II to 1965 Edition including addenda through Summer 1967 of ASME B&PV Code.
- CRDM nozzle material was hot rolled and annealed by B&W Tubular Products Division.
- CRDM nozzles were shrink fit into reactor vessel head penetration and welded with a J-groove weld with Alloy 600 filler

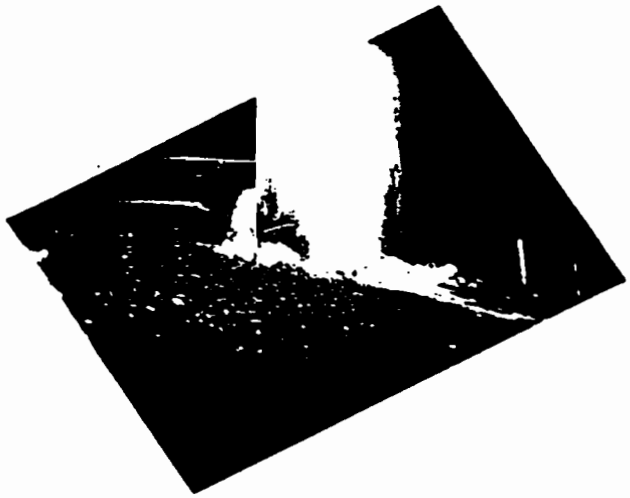
CRDM Nozzle Layout



Summary of Recent Cracking Incidents

- ONS-1:
  - All eight thermocouple nozzles contained flaws predominantly axial in orientation
    - Five nozzles identified as leaking
    - ID cracking observed on all eight nozzles
    - Cracking penetrated into all eight nozzle welds
  - CRDM nozzle 21 did not contain ID flaws
    - Flaws in weld material, predominantly axial/radial in orientation, identified as leak source
    - Flaw propagated through the weld and nozzle base material

ONS-1 RV Head Showing Boric Acid At CRDM Nozzle 21



Summary of Recent Cracking Incidents (Cont.)

- ONS-3:
  - Nine CRDM nozzles found leaking
    - Numerous axially oriented flaws identified
    - OD-initiated circumferential flaws (relatively deep and below the weld) identified on four nozzles
    - OD-initiated circumferential flaws (above the weld and up to through-wall) identified on two nozzles
    - Some weld cracking also identified

CRDM Nozzle #56

CRDM Nozzle #50

Summary of Recent Cracking Incidents (Cont.)

- ANO 1 CRDM nozzle 56 found leaking
  - No ID axially oriented flaws identified
  - One OD-initiated circumferential flaw below the weld that turned axial identified
  - Flaw propagated through the weld area along the nozzle OD

Visual Inspection
ANO 1

Investigations Performed
ONS 1 & 3

- Non-Destructive Examinations
- Metallurgical Examinations
- Analytical Evaluations

Non-Destructive Examinations

- Pre-Repair Inspections Performed
  - Visual inspections of all 69 CRDM nozzles
  - Dye Penetrant (PT)
  - Eddy Current Testing (ECT)
  - Ultrasonic Examination-Axial
  - Ultrasonic Examination-Circumferential

Visual Inspections

- Bare head inspections are performed through the modified openings in the head service structure
- Visual inspections are performed as part of each refueling outage for our response to GL 88-05 and 97-01
  - The same experienced system engineer performs these inspections
- Heads essentially clear of old boron deposits
- Amount of leakage from each leaking nozzle has been very small, which suggests, low leak rates
- No evidence of boric acid corrosion on top of head

Non-Destructive Inspections

- Dye Penetrant (PT) Inspection
  - Surface examination that looks at the weld surface area and the top 1 inch of the nozzle that projects down into the plenum of the head
  - Performed on suspected leaking CRDM nozzles
- Eddy Current (ECT) Inspection
  - Surface examination (plus 2 to 3 mm into the material) from the nozzle ID
  - Performed on suspected leaking nozzles
  - Checks a band 6 inches above the weld down to free end of nozzle
  - Later performed on additional nozzles, to address extent of condition
    - 8 Unit 1 CRDM nozzles
    - 9 Unit 3 CRDM nozzles

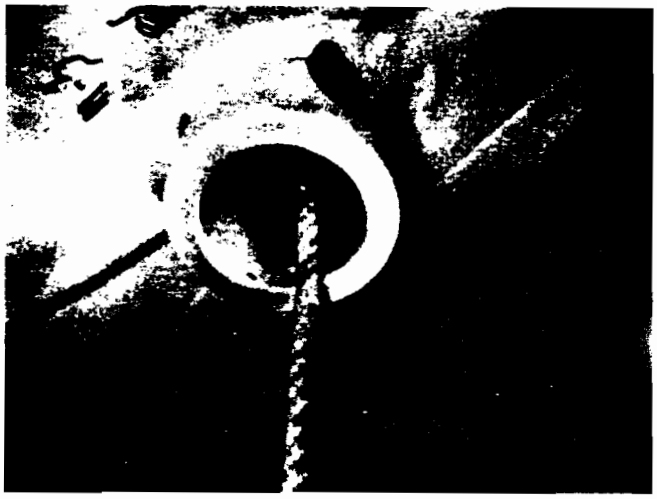
Non-Destructive Inspections

- Ultrasonic Examinations (UT) Axial
  - Volumetric examination to locate and depth size axial indications on both the nozzle inside diameter and the nozzle outside diameter
  - Performed on the suspected leaking nozzles and on additional nozzles to address extent of condition
    - 18 nozzles on Unit 3 inspected
- Ultrasonic Examinations (UT) Circumferential
  - Volumetric examination to detect the presence of circumferential cracking or indications and lack of bond
  - Performed on the suspected leaking nozzles and on additional nozzles to address extent of condition
    - 18 nozzles on Unit 1 (lack of bond)
    - 18 nozzles on Unit 3 (circumferential)

CRDM Nozzle #11

CRDM Nozzle #23

CRDM Nozzle #56



ONS 3: Summary Nozzle Indications and Characterization

- Total of 48 indications in the nine leaking CRDMs
  - 39 are axial and located beneath the weld at the uphill and downhill
  - 16 indications thru wall (39%), all are axial, and occur on 6 of 9 nozzles
- Confirmed two (2) above the weld circumferential cracks
  - Nozzle 56 crack was thru wall
  - Nozzle 50 except for pin hole indications on ID was not thru wall
  - Inspection and metallurgical results indicate the circumferential cracks were O.D. initiated.
- Unit 3 CRDMs extent of condition inspections (9 additional nozzles):
  - Cluster indications above and/or below the J groove weld.

Circumferential Cracks Above Weld

- Discovered during post weld repair NDE of Nozzles 50 & 56
- Circumferential cracks followed the weld profile contour and were O.D. initiated.
- Both ECT and UT inspections identified indications in these areas but were dispositioned as crazed cracks with unusual characteristics
- The original NDE characterization for nozzles 50 and 56 subsequently changed.
- This change in interpretation of the NDE signals is related to the flaw orientation with respect to the sound beam of the UT search units.
- Actions taken as a result of this discovery were:
  - All Unit 1 and 3 ECT and UT data re-reviewed applying the LLs
  - EPRI NDEC led an independent review of ONS 1 & 3 data to confirm results and findings

Metallurgical Examinations

- T/C nozzle specimen (2) from Unit 1
- CRDM #21 182 weld filler material boat sample from Unit 1
- CRDM nozzle end pieces (7) from Unit 3
- CRDM nozzle 56 circumferential crack boat sample, Unit 3

Unit 1: Summary Results of Metallurgical Examinations

- T/C Nozzles:
  - Cracks are intergranular and branched
  - Cracks are axial and radial in orientation
  - Material appears to be typical of mill annealed Alloy 600 with some evidence of cold working on both the OD and ID surfaces
  - Microstructure mixed with both intra and intergranular carbides
  - Microstructure characterized by small clusters of small grain with some large grains; Grain size ASTM 7-8
  - No indication of aggressive chemical species on the crack face
  - PWSCC was the primary mechanism for crack propagation

Unit 1: Summary Results of Metallurgical Examinations

- CRDM Nozzle 21:
  - Crack in weld was completely interdendritic
  - No conclusive evidence of manufacturing defects in the original weld
  - Crack in weld was connected to a branched intergranular crack in the nozzle wall
  - Qualitative comparison of boat sample to a 182 weld pad confirmed alloy type material, as expected
  - PWSCC was the primary mechanism for crack propagation in the CRDM weld and housing

Unit 3: Summary Results of Metallurgical Examinations

- CRDM Housing Material Specimen:
  - Microstructure of all nozzle materials very similar and typical for mill annealed Alloy 600. Grain size is ASTM 4.
  - Grain boundaries contain a semi-continuous carbide decoration
  - No ghost grain boundaries or segregated carbide clusters
  - All cracks in the samples were intergranular with slight branching
  - Micro-hardness survey across the thickness shows a range from about Rb 80 at the ID to Rb 95 at the OD
  - Several nozzles exhibited cracks originating at free end of nozzle
  - All cracks are stress corrosion cracks with PWSCC as the primary mechanism for crack propagation

Unit 3: Summary Results of Metallurgical Examinations

- CRDM 56 Boat Sample (Circ Crack):
  - Boat sample in the area of circ crack that was found above the weld after the weld repairs were completed
  - Boat sample contained a face of the circ crack along with 3 small axial cracks that intersect the circ crack
  - Section through the axial crack confirms crack is totally intergranular with small intergranular branches
  - Scanning electron microscopy of the circ crack face revealed only intergranular morphology.
  - There are no tears or other indications of the origin of the circ crack
  - Circ crack is indicative of PWSCC

Correlation of Observed Crack Locations with FE Stress Analysis

- Cracks are:
  - predominantly axial and located on the uphill and downhill sides of the nozzle
  - most initiate on the OD of the nozzle
  - circumferential cracks found below and above the weld, at the weld toe on the uphill and downhill sides of the nozzle

Correlation of Observed Crack Locations with FE Stress Analysis

- Stress analysis (residual + operation) preliminary results:
  - Hoop stresses exceed axial stresses at most locations which suggests axial cracking would be expected. This is consistent with observed field conditions
  - Axial stresses are higher on the uphill side of the nozzle relative to downhill side of nozzle. Field observed locations of the above the weld circumferential cracks align with this analysis prediction.
  - Microhardness measurements suggest the material yield strength is significantly higher on outside of nozzle than on the inside. The high outside yield strength may explain the preferred OD cracking

Oconee Repairs

- Repairs performed in accordance with 1992 Section XI of ASME Code, applicable Code Cases, and NRC approved alternatives, as required
- Removed flaws from both weld material and nozzle base material for Units 1 & 3
  - Automated weld process to apply protective layer over J groove weld
- Automated repair method used for Unit 2 removed cracked nozzle material and established new pressure boundary location. Cracks left in remaining J-groove weld

CRDM Nozzle #50

ANO 1 Repair

- Embedded Flaw Repair
  - OD axial flaw removed down to the butter
  - Weld repaired, isolating remaining flaw above the weld from the environment
  - Peened repair area
- Post-repair UT to confirm remaining flaw did not grow during repair process

Industry Response

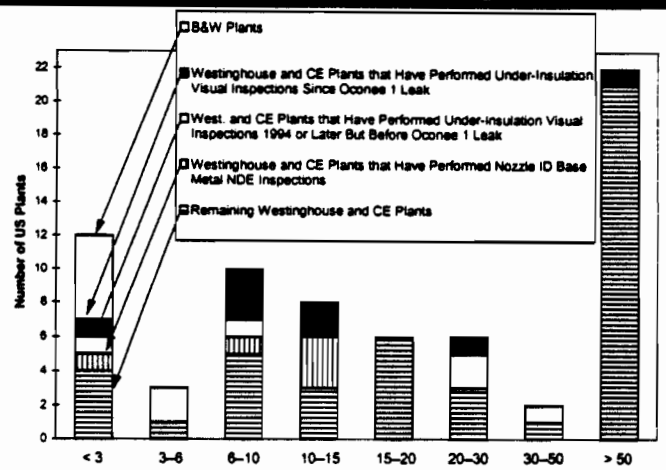
Industry Response Organization

- Integrated effort is being coordinated through
  - EPRI Materials Reliability Project - Alloy 600 ITG
    - NEI - Regulatory Interface
    - Committees Under Alloy 600 ITG
      - Assessment
      - Inspection
      - Repair/Mitigation
    - Owners Groups
- Work is being performed by
  - Utilities
  - NSSS Vendors
  - Contractors

MRP Interim Safety Assessment

- Interim Safety Assessment Submitted May 18, 2001
- Developed a Histogram of Time for Each Unit to Reach the Equivalent Time at Temperature as ONS 3 (normalized to 600F)
  - Sorted plants into bins, <3 EFPY, 3-6 EFPY, 6-10 EFPY, etc.
- Recommended Plants <10 EFPY from ONS 3 with Fall Outages perform visual inspections
  - Capable of detecting small amounts of Boron similar to ONS & ANO

Time-Temperature Histogram Chart in MRP-44 Part 2 Interim Safety Assessment



Effective Full Power Years (EFPYs) from 3/1/2001 until Reaching Oconee 3 EFPYs at Time of its Spring 2001 Outage, Normalized For Differences in RV Head Temperature Using the Arrhenius Relationship and a Standard Activation Energy of 50 kcal/mole

MRP Interim Safety Assessment

- Bases for No Significant Near-term Impact on Plant Safety:
  - The Three Oconee Units and ANO-1 Are Among the Lead Units in the US Based on Time at Temperature
  - Leaks Were Found by Careful Visual Inspections
  - Structural Integrity Evaluations Showed the Nozzles and Welds Were Well Within Required Margins
  - Leakage Should Also Be Detectable in Other Plants
  - Several Other Lead Units With Long Operating Times and High Head Temperatures Had Already Performed Inspections From Above and Below the Head Without Any Significant Findings
  - A CRDM Nozzle Ejection Is an Analyzed Event in Plant FSARs
  - Existing Symptom Based EOPs and Operator Training Adequate

NRC Questions

- NRC identified several questions on May 25, 2001:
  - Leak detection
    - Effect of initial interference fit on leak detection
  - Time-temperature histogram
    - Effect of activation energy on predictions
    - Benchmarking against foreign plant inspections
    - Basis for ten year inspection criterion
  - Growth rate of circumferential cracks
    - Time until Oconee 3 would have reached allowable flaw size
    - Effect of crack growth rates on histogram
  - Loose parts
  - Risk assessment

NRC Questions

- NRC Documented Those and Asked Additional Questions on June 22, 2001:
  - Photos of visual inspections performed at other units
  - Inspection Capabilities
    - Ability to Perform Volumetric NDE
      - Nozzles for ID/OD Flaws
      - J-groove Welds
    - Estimate of Number, Time, Other Costs to Perform Volumetric and Visual Inspections by 1/1/2002
      - During Scheduled Outage
      - During Unscheduled Outage

Safety Assessment Status

- The Interim Safety Assessment was prepared to demonstrate safety of operating plants
- Additional effort is ongoing in several areas
  - Analysis associated with the Final Safety Assessment
  - Visual inspections of the reactor vessel top head surface for plants coming down for Fall 2001 refueling outages
  - Research into improved inspection and repair technology
  - Risk assessment
- Results will be factored into the Final Safety Assessment

Leakage Detection

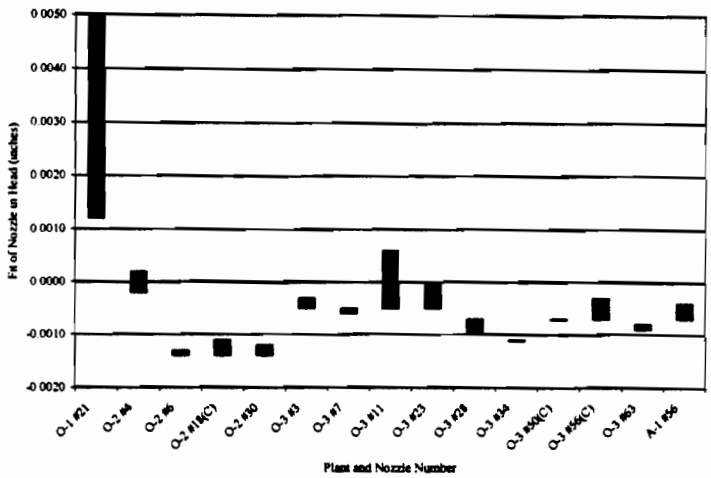
Leakage Detection

- Oconee and ANO-1 detected leakage, but
  - Some other plants have greater interference fits (see Table 3-2 of Interim Safety Assessment)
- Leakage should be detectable at most other penetrations given similar cracks
  - Only minor craze cracking was found in NDE examinations of 17 additional "non-leaking" Oconee 1 and 3 CRDM nozzles. This supports appropriateness of visual inspections for detection of through-wall cracks in CRDM nozzles
  - Interference fits at other plants are only slightly larger than Oconee and ANO-1
  - Further experience has shown that it is difficult to prevent leakage of 2,250 psi water without roll, hydraulic or explosive expansion or use of a sealant

Leakage Detection
Actual Fits at Oconee and ANO-1

- Fabrication records for Oconee 1, 2, and 3 and ANO-1 vessel heads have been reviewed
- The following measurements were taken
  - ID of the hole in the vessel head at the top and bottom of the interference fit region
  - OD of the nozzle
- Results for the 14 leaking CRDM nozzles at Oconee 1, 2, and 3 and ANO-1 are shown on next slide
  - One nozzle had a clearance fit (gap)
  - The remaining nozzles had at least one end within the specified diametral interference range of 0.0005 - 0.0015 inches. Three of the four leaking ONS 2 nozzles had interference fits of 0.0014 inches on one end and at least 0.0011 inches on the other.

Leakage Detection
Actual Fits at Oconee 1, 2, and 3 and ANO-1



Leakage Detection
Effect of Operating Conditions on Fit

- Differential thermal expansion has only a small effect, increasing the initial interference fit by <0.0014"
- The change in fit under operating conditions is primarily due to pressure dilation of the vessel head
- For the example, the change in diametral fit due to pressure dilation is approximately
  - ΔD = 0.00402" - 0.00048" = 0.0035"
  - The hole will open up further when the effect of reduced effective modulus due to the effect of multiple nozzles is considered
- Therefore annular gaps are expected for most CRDM nozzles under operating conditions

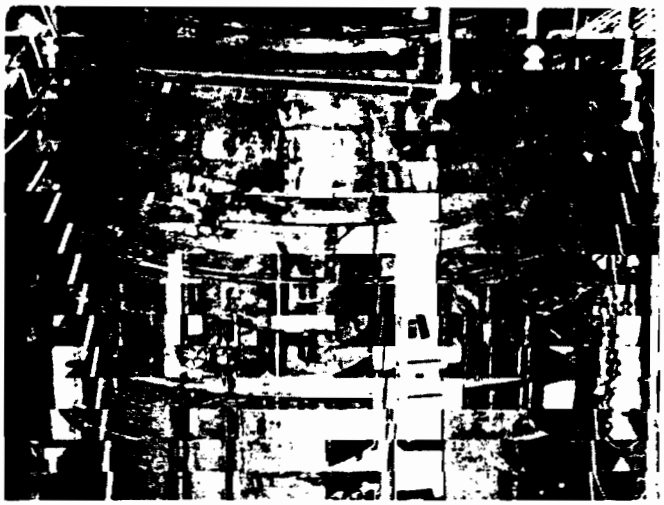
Leakage Detection Other Effects on Fit

- Finite element analyses show that outer row CRDM nozzles displace laterally and become slightly ovalized in the vessel head as a clearance opens up under operating conditions
  - The displacement and ovalization reduce the leak path at some locations and increase the leak path at other locations
  - The net effect is to create a spiral flow path which has less resistance than a uniform annular gap
- Finite element analyses also show a minor (~20%) increase in ovality for peripheral CRDMs from flange tensioning and rotation

Visual Inspections Spring 2001

- Several other plants performed visual inspections during Spring outages
  - Robinson 2
  - Salem 1
  - Farley 2
  - Prairie Island 1
  - McGuire 1 (partial)
  - SONGS 3 (partial)
- Heads reasonably free of masking boric acid deposits
- No evidence of leakage found

Visual Inspection
Salem 1



Time Temperature Histogram

Time-Temperature Histogram Background

- The time-temperature model groups plants according to the time (EFPY) required for each unit to reach the equivalent effective time at temperature as Oconee 3 at the time the above-weld circumferential cracks were discovered in February 2001
- The reference date for the time-temperature assessments is March 1, 2001
- The industry standard activation energy of 50 kcal/mole for PWSCC initiation in Alloy 600 material was used to normalize plant operating time to a head temperature of 600°F

Time-Temperature Histogram Effect of Activation Energy (cont.)

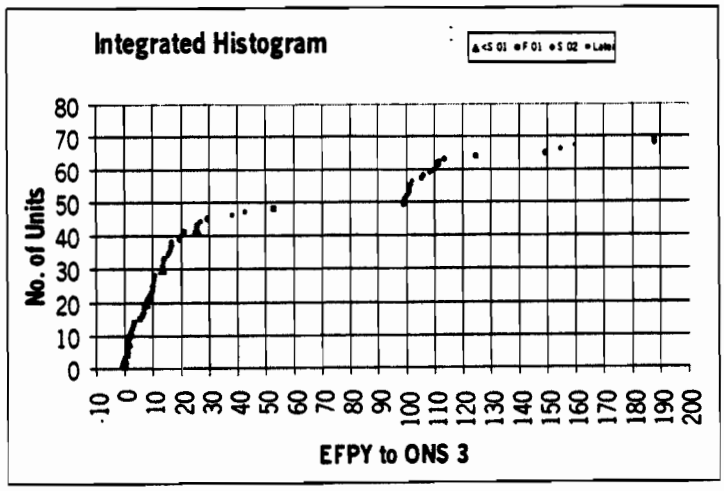
- A sensitivity study for the results of the plant assessments was performed
- The effect is small, as shown below:

Activation Energy	*Assessment Groups*							
	< 3 EFPYs	3–6 EFPYs	6–10 EFPYs	10–15 EFPYs	15–20 EFPYs	20–30 EFPYs	30–50 EFPYs	> 50 EFPYs
50 kcal/mole	12	3	10	8	6	6	2	22
40 kcal/mole	12	4	14	9	4	3	2	21

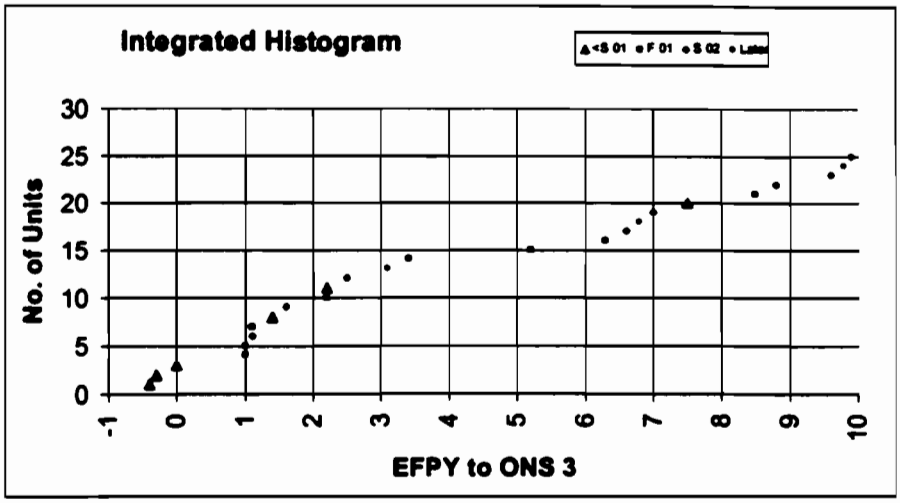
Time-Temperature Histogram Ten-Year Period

- 10 Year Period for Near-Term Inspection
  - The ten year period for recommending visual inspections of the top of the vessel head for small amounts of leakage similar to that observed at Oconee and ANO-1 was selected to provide some margin for uncertainties
  - Encompasses 25 units
  - All but two will have outages by Spring '02
  - The ten year period will be re-assessed based on results of upcoming outages

Time-Temperature Histogram



Time-Temperature Histogram



Circumferential Crack Growth Growth Rate in Annulus Environment

- Data are available from 5 sources for carefully controlled PWSCC tests of Alloy 600 and 182, using PWR conditions
- OD initiated cracking requires the presence of water or steam, so a pressure boundary leak is necessary
- The crevice region could contain some Oxygen from the containment atmosphere, but at temperature this Oxygen would be quickly consumed by reaction with the low alloy steel nearby
- This reaction, plus the extremely tight fit and the distance to the OD of the head, make a high Oxygen environment unlikely

Crack Growth

Circumferential Crack Growth Growth Rate in Annulus Environment

- Since the fluid will contain lithium hydroxide and boric acid, it will likely be similar to a controlled PWR environment
- Comparison of BWR and PWR crack growth rates for Alloy 600 and 182 shows that, at a given temperature, the growth rates are comparable
- Temperature is a stronger variable than environment for these materials
- MRP has scheduled an international expert panel to assess crack growth rates
  - Initial meeting in August

Circumferential Crack Growth Margin for Oconee 3 Cracks

- Two Oconee 3 nozzles were cracked approximately 165°
- Stress analyses show that cracks initiated in a high stress region and propagated into a lower stress region
- The remaining time for Oconee 3 circ cracks to reach ASME Code allowable ligament (safety factor of 3) was estimated to be 4-5 years, based on the modified Peter Scott model and also by assuming the maximum crack growth measured in lab
- Efforts are underway to refine the stress intensity calculations in the nozzle in the intact and cracked conditions

Loose Parts & Risk Assessment

Loose Parts

- The potential for, and consequences of, loose parts in B&W designed plants such as Oconee and ANO-1 was described to the NRC on April 12, 2001
- Creation of loose parts was deemed unlikely
- Worst postulated condition is a single stuck rod
- While analyses for other plant designs have not been completed, results are expected to be similar
- Loose parts analyses will be included in final report

Risk Assessment

- Risk calculations are in process now
- The effort includes interaction with all PWR vendors and others to ensure applicability to all plants
  - Consistent with past approaches
- Staff has conservatively estimated CCDP about 10-3, assuming rod ejection, but probability of ejection event likely to be a few orders of magnitude less than 1 for all plants

Summary & Ongoing Activities

Summary

- Near Term Conclusion:
  - Axial cracks alone in CRDM nozzles do not impact plant safety
    - Bounded by previously submitted Safety Assessments (1993/94)
    - But through wall axial cracks can be a precursor to circumferential cracking
  - There is reasonable assurance that PWRs do not have circumferential cracking that would exceed structural margin
    - Oconee and ANO-1 in highest grouping based on effective time-at-temperature
    - Leaks discovered by careful visual inspection of top head surface
    - Volumetric examination of other nozzles found only minor craze cracks
    - Leaks discovered with significant structural margin remaining
    - Several other plants in highest groupings have no evidence of leakage

Schedule

- Revised Inspection Recommendations - July-August
- Expert Panel on Crack Growth - First Meeting 8/01
- Inspections during Fall 2001 outages
- Final RPV Penetration Safety Assessment - 12/01
- Reassessment of Inspection Recommendations - 2/02

Other Ongoing MRP Activities

- Risk Assessments
- Probabilistic Fracture Mechanics
- NDE Demonstration
  - Block Design and Fabrication
  - Technique Development and Demonstration
- Information and Training Package for Visual Examination
- Flaw Evaluation Guidelines
- Review of Repair and Mitigation Strategies

United States Nuclear Regulatory Commission

PERSPECTIVES GAINED FROM INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) PROGRAM

by

Alan Rubin, Section Chief
Probabilistic Risk Analysis Branch
Division of Risk Analysis and Applications
Office on Nuclear Regulatory Research

Presentation to ACRS

July 12, 2001

8

OUTLINE OF PRESENTATION

- Methodologies (seismic, fire, human error)
- IPEEE-related unresolved/generic safety issues
- Conclusions/further actions

OUTLINE OF PRESENTATION TO ACRS SUBCOMMITTEE ON RELIABILITY AND PRA (JUNE 22, 2001)

- Introduction
- IPEEE seismic perspectives
- IPEEE fire perspectives
- High winds, floods and other (HFO) external events
- IPEEE-related generic safety issues (GSIs/USI)
- Uses of IPEEE information
- Conclusions and observations

METHODOLOGIES - SEISMIC

- Seismic margin vs. seismic PRA
- Human error
- Surrogate elements
- Uniform Hazard Spectrum (UHS)
- Simplified fragilities
- Soil evaluation

METHODOLOGIES - SEISMIC (cont.)

- Industry/NRC activities
  - "External Events PRA Methodology Standard" (American Nuclear Society)
  - Proposed Revision 2 to Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants" (NRC)

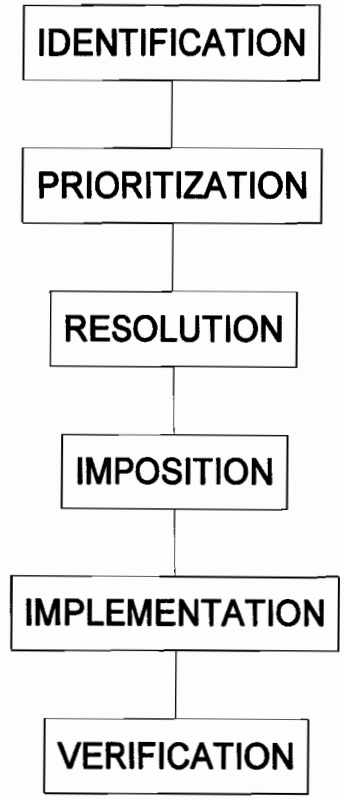
METHODOLOGIES - FIRE

- FIVE vs. fire PRA
- Fire PRA Implementation Guide
- Human error (recovery actions)
- Severity factors
- Circuit analysis
- Fire modeling
  - Multi-zone fire analysis
  - Fire barrier reliability
- Electric panel fires
- Effectiveness of fixed detection and suppression
- Self-ignited cable fires

METHODOLOGIES - FIRE (cont.)

- Industry/NRC activities
  - Fire Risk Research Program (NRC)
    - Fire risk assessment methods development
    - Fire model benchmarking and validation
    - Fire risk requantification study
  - HRA research (NRC)
  - Supplemental guidance in response to NRC's generic RAIs on Fire PRA Implementation Guidance (EPRI)
  - Standard on fire PRA (American Nuclear Society)

Generic Issue Process (from NUREG-0933)



IPEEE-RELATED USI/GSIs

- USI A-45, GSI-57, GSI-103, GSI-131, GSI-147, GSI-148, and IPEEE-related aspects of GSI-156
  - These issues are already considered resolved. IPEEE review is for verification. No further generic action necessary.
- Fire Risk Scoping Study Issues
  - Not part of the generic issues program, although some issues became generic issues (GSI-57, GSI-147, and GSI-148).

IPEEE-RELATED USI/GSIs (cont.)

- GSI-172, Multiple System Responses Program (MSRP)
  - This issue is still considered "open."
  - IPEEE verified 80% of the plants have adequately addressed the IPEEE aspects of this issue.
  - Higher verification percentage than usual for generic issue implementation.
  - A resolution package for this issue will be generated and submitted for ACRS review, in accordance with standard generic issue procedures.

CONCLUSIONS/FURTHER ACTIONS

- IPEEE program successful in meeting intent of Supplement 4 to Generic Letter 88-20
- IPEEE reviews verified resolution of large majority of IPEEE-related USI/GSIs
  - Need for additional actions/assessments on issues for some plants will be determined separately from IPEEE program.
- Public comments due July 31, 2001, on Draft NUREG-1742
- Issue final NUREG-1742 (Oct. 2001)

United States Nuclear Regulatory Commission

IPEEE-RELATED UNRESOLVED SAFETY ISSUE (USI) AND GENERIC SAFETY ISSUES (GSIs)

by

John N. Ridgely
Probabilistic Risk Analysis Branch
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

Presentation to ACRS

July 12, 2001

Licensees were specifically requested to address the following issues

- USI A-45, "Shutdown Decay Heat Removal Requirements"
- GSI-103, "Design for Probable Maximum Precipitation"
- GSI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants"
- GSI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment"
- Sandia Fire Risk Scoping Study (FRSS) issues

IPEEE information could be used to verify

- GSI-147, "Fire-Induced Alternate Shutdown/Control Room Panel Interactions"
- GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness"
- GSI-156, "Systematic Evaluation Program" (SEP)
- GSI-172, "Multiple System Responses Program" (MSRP)

USI/GSI Staff Review Evaluation Process

- The licensee's IPEEE is complete with regard to USIs and GSIs coverage.

- The licensee's assessment demonstrated an in-depth knowledge of the external events aspects and plant characteristics relevant to the issues discussed.

- The licensee's assessment results are reasonable given the design, location, features, and operating history of the plant.

An issue is thus considered adequately verified if no potential vulnerabilities associated with its related concerns were identified in the submittal, or plant-specific improvements to eliminate or reduce the significance of the identified potential vulnerabilities were implemented at the plant.

Generic Safety Issues Addressed in the IPEEE Program

Generic Safety Issue (GSI)	Area[1]	USI/GSI	FRSS	GSI-156	GSI-172	Remark[2]
Shutdown Decay Heat Removal Requirements	S,F	USI A-45				EX
Potential Seismic Interaction Involving the Movable In-Core Mapping System	S	GSI-131				C
Effects of Fire Protection System Actuation on Safety-Related Equipment	S,F	GSI-57	X		X	P
Fire-Induced Alternate Shutdown/Control Room Panel Interaction	F	GSI-147	X			C
Smoke Control and Manual Fire-Fighting Effectiveness	F	GSI-148	X			P
Seismic/Fire Interactions	F		X		X	C
Adequacy of Fire Barriers	F		X			C
Effects of Hydrogen Line Ruptures	S,F				X	C
Settlement of Foundations and Buried Equipment	S			X		P
Dam Integrity and Site Flooding	S,HFO			X		C
Seismic Design of Structures, Systems, and Components	S			X		C
Common Cause Failures Related to Human Errors	S,F				X	EX
Non-Safety-Related Control System/Safety-Related Protection System Dependencies	S,F				X	EX
Effects of Flooding and/or Moisture Intrusion on Non-Safety-Related and Safety-Related Equipment	F,HFO				X	EX
Seismically Induced Spatial and Functional Interaction	S				X	C
Seismically Induced Flooding	S				X	C
Seismically Induced Relay Chatter	S				X	C
Evaluation of Earthquake Magnitudes Greater than Safe Shutdown Earthquake	S				X	C
Design for Probable Maximum Precipitation	HFO	GSI-103				C
Site Hydrology and Ability to Withstand Floods	HFO			X		P
Industrial Hazards	HFO			X		C
Tornado Missiles	HFO			X		C
Severe Weather Effects on Structures	HFO			X		C
Design Codes, Criteria, and Load Combinations	S,HFO			X		C
Shutdown Systems and Electrical Instrumentation and Control Features	F			X		EX

[1] S=seismic, F=internal fires, HFO=high winds, floods, and other external events

[2] C=issue covered by IPEEE; EX= only external event-related aspects of issue covered; P=partially covered (refer to specific section of the text for details)

USI A-45, "Shutdown Decay Heat Removal (DHR) Requirements"

- Objective to determine whether the decay heat removal function is adequate and wherever cost-beneficial improvement(s) could be identified.
- Components needed defined in NUREG-1289 (backfit analysis).
- IPE (NUREG-1560) performed PRA for systems and components, including those needed for DHR (internal events).
- IPEEE considered how ***external*** events (seismic, fire, and HFO) could adversely effect systems and components needed for DHR.

USI A-45 DHR (Continued)

Seismic Findings:

- Seismic PRAs included DHR systems and components.
- Seismic Margin Analyses (SMA) included DHR systems and components in their safe shutdown equipment list (SSEL).
- For SMA, each component's high confidence of low probability of failure (HCLPF) value was determined.
- Seismic walkdown information was used.
- Weaknesses were identified (e.g., weak anchorage of RHR heat exchangers) and plant improvements implemented.
- No vulnerabilities were found.

USI A-45 DHR (Continued)

Fire Findings:

- Licensees performed a fire PRA which included DHR systems and components (IPE model).
- Licensees indicated no DHR vulnerability by:
  - Qualitative screening - a fire in an area neither initiates an event nor causes the loss of safe shutdown functions;
  - Quantitative screening - contribution from a fire area was less than 10-6/ry.
  - Remaining fire areas were reviewed on a case-by-case basis to ensure at least one method of safe shutdown (frequently using Appendix R systems) and DHR was available.
- Fire walkdowns information was used.
- No vulnerabilities were found.

USI A-45 DHR (Continued)

HFO Findings:

- Safety-related equipment (including Appendix R equipment) is protected from high winds, tornadoes, and tornado-generated missiles.
- External flooding-induced failure prevented by watertight structures, leakage would be limited to prevent damage, or equipment operable submerged; otherwise function provided by alternate system.
- Other external events (e.g., lightening, chemical hazards, airplane crashes) were found to be insignificant contributors to core damage frequency.
- Walkdown information was used.
- No vulnerabilities were found.

USI A-45 DHR (Concluded)

Conclusion:

- All plants have provided adequately information to verify USI A-45.
- All plants have identified at least one method of removing decay heat.
- No vulnerabilities were found.

GSI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment"

- Objective is to evaluate potential risks from seismically induced fire plus seismically induced suppression diversion and seismically induced actuation of the fire protection system (FPS).

- One aspect ***not*** addressed as part of the IPEEE is related to other potential damaging effects, i.e., smoke and fire suppressant damage to other equipment not directly affected by the fire.

- Also discussed as part of the Sandia Fire Risk Scoping Study and as part of GSI-172.

- Walkdowns were to assess whether:
  - actuation of FPS would spray safety-related equipment
  - some protective measures, if needed, could provide protection of safety-related equipment.

GSI-57 (Continued)

Findings:

- Some submittals noted the plant's FPS was designed per Category II/I criteria.
- Pre-Action Type: requires two diverse actions for initiation, smoke detector to open a supply valve and fusible link in the sprinkler head.
- Deluge Type: relies on spatial relationship between the FPS and safety-related components, seals, drainage systems.
- CO_2 or Halon: reviewed for potential effects on personnel (e.g., control room operators) and equipment (e.g., diesel generators operation).

GSI-57 (Concluded)

Conclusions:

- Licensees concluded the impact was negligibly small.
- No plant vulnerabilities were identified.
- All but four plants have provided adequate information to verify this issue.
  - One plant provided no information.
  - Three plants provided partial information.

GSI-103, "Design for Probable Maximum Precipitation" (PMP)

- Objective is to evaluate potential effects of new PMP criteria which might increase potential site flooding levels and roof ponding loads.

- Used revised NOAA's hydrometeorological reports.

- Related to seismic only with respect to potential failure of upstream dams, levies, and ponds and their potential for increased site flooding.

- Walkdowns looked for:
  - potential water ingress routes into structures, including doors and penetrations,
  - roof drains, including plugging,
  - roof scupper capacities, and
  - plant grading, including drainage features.

GSI-103 PMP (Continued)

Findings:

- Typically, roofs can withstand the additional loads because the excess rainfall overflows the roof parapets.
- In some cases, scuppers were installed in the parapets.
- To credit roof drains, licensees referred to procedures to periodically inspect the roof drainage system for potential blockage.
- Typically, site flooding from PMP effects on nearby rivers and streams (potential dam and levy failures) did not adversely affect the plant.
- If flooding could adversely affect the plant, plant changes were made (sand bags, timely shutdown).

GSI-103 PMP (Continued)

- Walkdowns did identify water ingress paths that licensees adequately addressed.
- Site drainage adequately removed very intense local precipitation.
- Potential flooding conditions were:
  - insignificant water accumulation,
  - significant water accumulation, but no equipment flooding, or
  - components operate submerged.

GSI-103 PMP (Concluded)

Conclusions:

- Original design and construction of the plants included sufficient margin to allow for variations of up to two to three times the original design basis PMP without adversely impacting safe operation of the plant.
- No plant vulnerabilities were identified.
- One plant (Salem) installed new penetration seals between the service and auxiliary buildings. Reduced estimated CDF from external floods from 1E-4/ry to 1E-7/ry.
- All but two plants used revised PMP data.
- All but three plants provided adequate information to verify all aspects of GSI-103.

GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness"

- Objective is to determine whether licensees have adequately considered the potential effects of smoke. Buildup of smoke could:
  - reduce manual fire-fighting effectiveness,
  - potentially damage equipment from misdirected spray,
  - hamper operator's ability for safe shutdown due to MCR abandonment and use of alternate shutdown capability,
  - initiate fire protection system resulting in diversion from fire area and potential equipment damage. (GSI-57)

- One aspect ***not*** addressed as part of the IPEEE is the potential for smoke to cause equipment to be damaged or degraded. This is addressed in NUREG/CR-6597 (1/01).

GSI-148 (Continued)

- Also discussed as part of the Sandia Fire Risk Scoping Study and GSI-57.

Findings:

- 65% credited manual fire-fighting actions.
- 15% did not explicitly discuss, but could be evaluated based on review of the FRSS issues.

  These addressed:

  - delays in manual actuation of suppression systems,
  - delays in locating the fire, after fire brigade arrival,
  - time to extinguish fire, and
  - fire brigade training, including timing records, live fire/smoke exercises, and plant configuration simulation.

- 20% took no credit for manual fire-fighting activities.

  - Conservative assumption from PRA standpoint (i.e., higher CDF estimate).

GSI-148 (Continued)

- However, does not consider potential effects of:
  - misdirected spray,
  - breached fire barriers (leading to spread of smoke, fire, or both to adjacent fire areas).

- Even those that took no credit discussed fire brigade training, simulation exercises, equipment, and timing.

- Effects of corrosion, buildup of soot, or other combustion products discussed by few.
  - some stated these were long-term issues resolved by corrective maintenance,
  - some limited discussion to use of SCBA and portable ventilation equipment,
  - Two addressed potential toxic and corrosion effects.

GSI-148 (Concluded)

Conclusions:

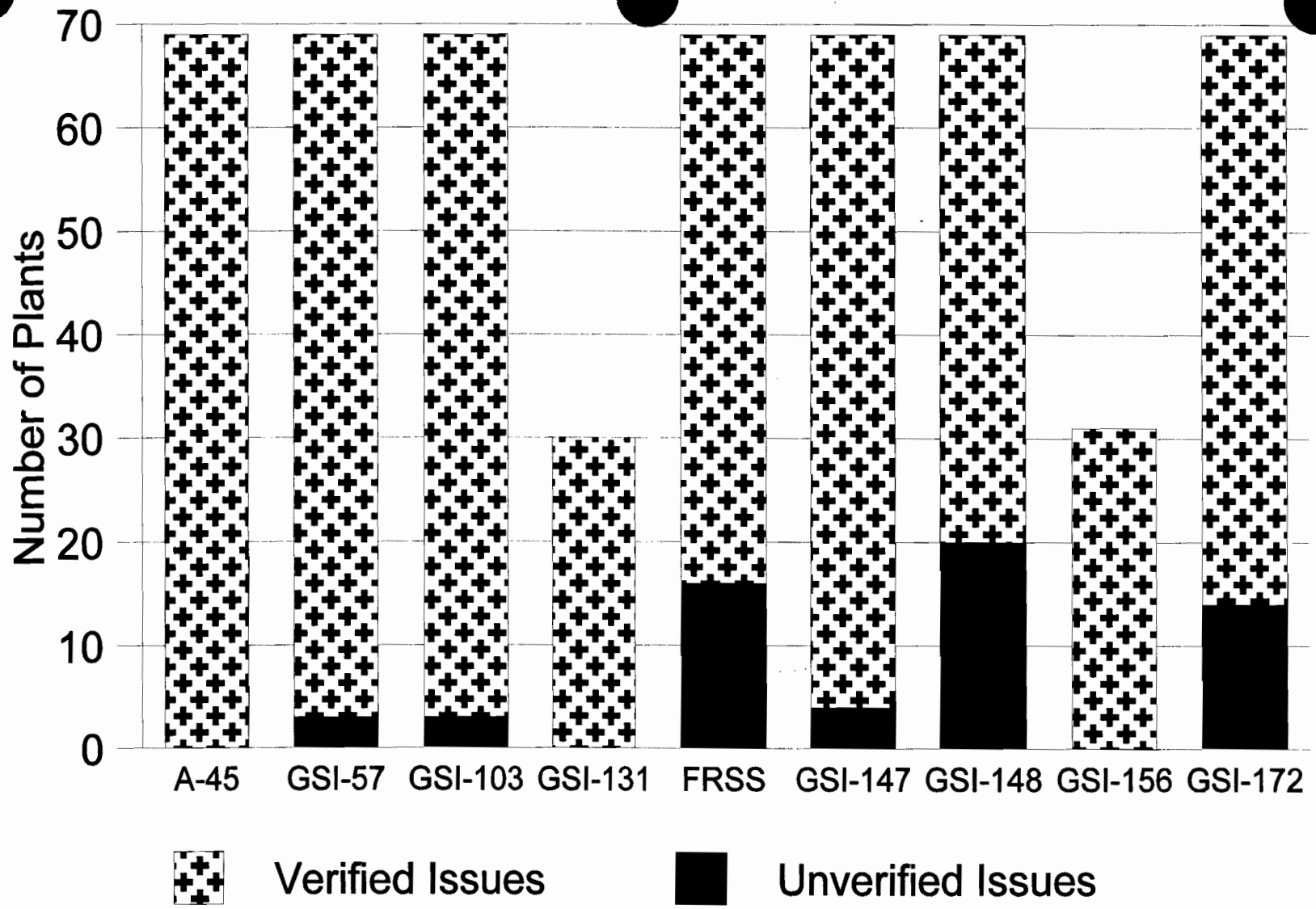
- No plant vulnerabilities were identified.
- 71% of the plants provided adequate information to verify this issue.
- 25% of the plants provided part of the information needed to verify this issue.
- 4% of the plants did not provide information to verify this issue.

Summary and Conclusions

- 31 IPEEE-related unresolved safety issues and generic safety issues (issues and sub-issues).
  - 9 explicitly discussed in Supplement 4 to Generic Letter 88-20 and NUREG-1407.
    (USI A-45, GSI-57, GSI-103, GSI-131, and 5 FRSS issues)
  - 22 issues were not explicitly discussed in the Generic Letter or NUREG-1407.
    (GSI-147, GSI-148, GSI-156 [9 issues], and GSI-172 [11 issues].)
- Major achievement is verification of a large majority of these generic issues
  - 44 licensees provided sufficient information to verify all 31 USIs and GSIs.
  - 25 submittals had one or more generic issue(s) or sub-issue open or only partially verified.

Summary and Conclusions (Continued)

- Verified:
  - 100% USI A-45, GSI-131, and GSI-156
  - 95% GSI-57, GSI-103, and GSI-147
  - 80% GSI-172 and Sandia FRSS
  - 70% GSI-148
- For those issues not fully verified:
  - Potential "vulnerability" not missed.
  - Identified as "weakness" in plant-specific SER.
  - Need for additional plant specific actions or assessments to complete verification of these issues will be determined separately from IPEEE program.



Number of Unverified Unresolved and Safety Generic Issues by Plant
(not verified or partially verified)

(9)

Results of GSI-191 Parametric Evaluation

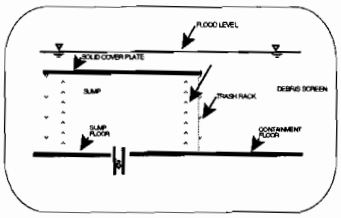
RES Contacts:
Mr. Michael Marshall, 301-415-5895
Mr. John Boardman, 301-415-6354

LANL Contacts:
Dr. Bruce Letellier, 505-665-5188
Dr. D.V. Rao, 505-667-5098

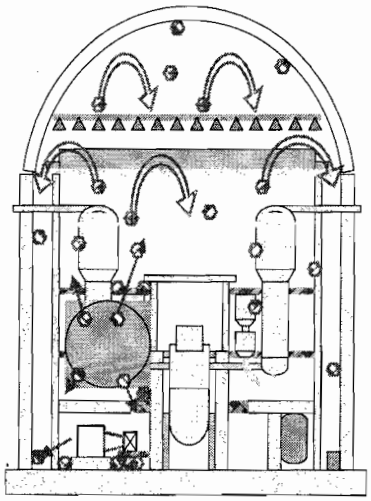
ACRS Presentation
Rockville, MD
July 12, 2001

Purpose of GSI-191 Study

- Determine whether debris accumulation on sump screens will cause loss of net positive suction head (NPSH) margin following a loss-of-coolant accident (LOCA).
- Determine if further action needs to be taken for pressurized water reactors beyond what was done during the resolution of Unresolved Safety Issue A-43.



Overview of GSI-191 Study



- Identify Debris Sources
- Estimate Amount of Debris Generated
- Estimate Amount of Debris Transported
- Estimate Head Loss Caused by Debris
- Compare Head Loss to NPSH Margin

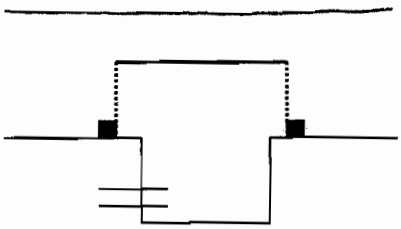
Compare Head Loss to Height of Pool

Rockville, MD
July 12, 2001

Definition of Sump Failure

- Fully Submerged Sump Screens

$$\Delta H_{screen} \geq NPSH_{margin}$$

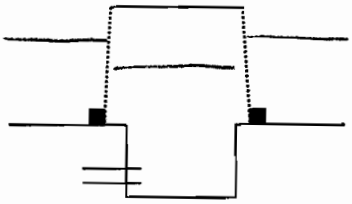


- Partially Submerged Sump Screens

$$\Delta H_{screen} \geq NPSH_{margin}$$

or

$$\Delta H_{screen} \geq \tfrac{1}{2} \text{ of pool height}$$



Rockville, MD
July 12, 2001

Purpose of Parametric Evaluation

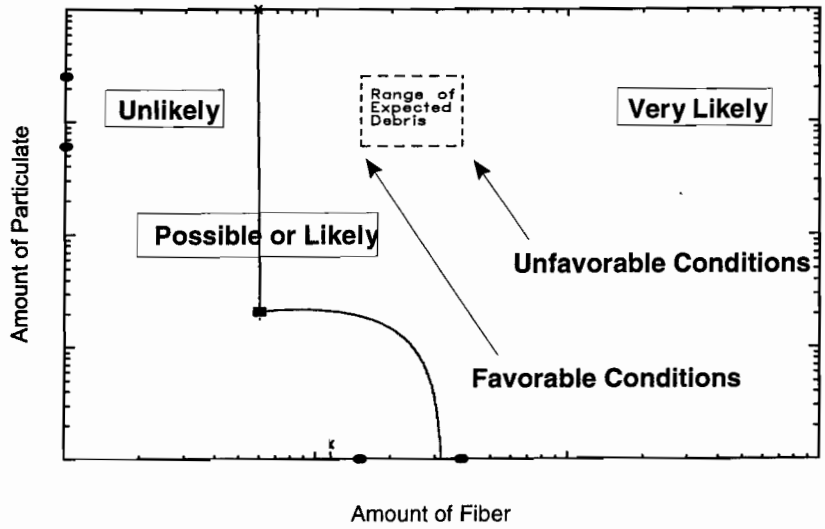
- Perform analyses that will demonstrate -generically - that debris accumulation will or will not cause loss of NPSH margin for ECCS pumps during recirculation given a LLOCA, MLOCA, or SLOCA
  - Analyses addresses debris generation, debris transport, debris accumulation, and the resulting head loss across the sump screen.
  - Analyses addresses variability in relevant plant features such as screen area, sump configuration, debris sources, etc.
  - Some relevant plant features could not be addressed such as debris location, containment configuration, etc.

Description of Parametric Cases

69 parametric cases

- Each case is based on an operating PWR unit
- Sump configuration based on survey responses
- Piping configuration based on one of two reference plants
- Type of thermal insulation based on survey responses
- Plant response to postulated LOCA based on MELCOR and RELAP calculations and survey responses
- NPSH margins based on licensee responses to GL 97-04
- Defined favorable and unfavorable conditions

Overview of Results



Overview of Results

Qualitative Grade	SLOCA	MLOCA	LLOCA
Very Likely	23	32	57
Likely	10	8	4
Possible	10	3	0
Unlikely	26	26	8
Total	69	69	69

The 69 parametric cases developed for this evaluation provide a reasonable representation of operating PWRs, so the results form a credible technical basis for making a determination of whether sump blockage is a generic concern for PWRs.

However, the parametric evaluations suffer from a number of limitations that make them ill suited for making a determination of whether a specific plant is vulnerable to sump failure.

Limitations of Parametric Evaluation

Most of the limitations to the calculations are due to lack of plant-specific information.

Many of the limitations have greater impact on SLOCA calculations than MLOCA or LLOCA

- Important information such as location of fibrous debris sources not included in analyses
- Effect of plant design on transport not addressed in analyses
- Possible changes in NPSH margin not addressed in analyses
- Cannot identify time failure occurs
- Mixture of actual, design, and licensing plant data

Insights From Parametric Evaluation

- Very little fibrous and particulate debris is needed to cause sump failure
  - Small NPSH margin
  - Small Sump Screen Area
- Most of parametric cases analyzed for LLOCA resulted in sump failure
- Some of the parametric cases analyzed for SLOCA resulted in sump failure

POWER UPRATES

NEED FOR STANDARD REVIEW PLAN SECTION FOR POWER UPRATE REVIEWS

ACRS 484th MEETING
JULY 12, 2001

OVERVIEW

- Background
- Current Guidance
- Potential Change to Review Processes
- Conclusions

BACKGROUND

- 12/1995 - Allegation on Maine Yankee Analyses
- 1/1996 - Order Limiting Power
- 4/1996 - Maine Yankee Lessons Learned Task Group Formed
- 7/1996 - Independent Safety Assessment (ISA)
- 10/1996 - ISA Report
- 11/1996 - EDO Direction to Address Recommendations in ISA Report
- 12/1996 - Maine Yankee Lessons Learned Report
- 4/1997 - NRR Committed to Develop a Standard Review Procedure

CURRENT GUIDANCE

- Approved GE Topical Reports (1991 - 1998)
- SEs for Monticello (BWR) and Farley (PWR) -"Templates" (1998)
- Applicable Standard Review Plan (SRP) Sections

CURRENT GUIDANCE (Example)

- Containment System Response
  - BWRs
    - Monticello SE Section 2.5
    - GE Topical Report Section 5.10.2
    - Applicable Subsections of SRP Section 6.2.1
  - PWRs
    - Farley SE Section 3.3
    - Applicable Subsections of SRP Section 6.2.1

POTENTIAL CHANGE

- Staff Requirements Memorandum Dated May 24, 2001

- Measurement Uncertainty Recapture Power Uprates

  - Review of Processes
  - New GE Topical Report

- Extended Power Uprates

  - First-of-a-Kind Applications (Duane Arnold, Quad Cities, Dresden)
  - Lessons Learned Workshop
  - Review of Processes
  - New GE Topical Report

- Will Issue Guidance via Regulatory Issue Summaries and/or External Website

CONCLUSIONS

- Sufficient Guidance Exists

  - Template Safety Evaluations
  - Approved Topical Reports
  - Current SRP

- Considering Explicitly Identifying the Monticello and Farley Safety Evaluations as "Template SEs " in the Project Managers' Handbook

- Processes Still Changing

- Resources Needed for Plant Specific Application Reviews

- Will Reevaluate Need for SRP Section in the Future

(11)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
484	13	13.1

Title **PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES**

Authors
JOHN T. LARKINS/SAM DURAISWAMY

List of Documents Attached

13

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	JOHN T. LARKINS/ SAM DURAISWAMY

G:PlanPro:ppmins.484.wpd
July 12, 2001

MINUTES OF THE
PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
TUESDAY, JULY 12, 2001

The ACRS Subcommittee on Planning and Procedures held a meeting Tuesday, July 10, 2001, in Room 2 B1, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 3:10 p.m. and adjourned at 4:40 p.m.

ATTENDEES

G. E. Apostolakis, Chairman
M. Bonaca
T. kress

ACRS STAFF

J. T. Larkins
J. Lyons
H. Larson
S. Duraiswamy
R. P. Savio
S. Meador
C. Harris

NRC Staff

I. Schoenfeld

DISCUSSION

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the July ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the July ACRS meeting are attached. Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the July 2001 ACRS meeting be as shown in the attachment.

2) Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through October 2001 is attached (pp. 7-10). The objectives are to:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

During this session, the Subcommittee discussed and developed recommendations on the items that require Committee decision, which are included in Section II of the Future Activities list (pp. 11-15).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee needs to consider the Subcommittee's recommendations on items listed in Section II of the Future Activities.

Since there were a large number of items scheduled for the September meeting, the Subcommittee has deferred several items to October 2001. Still, the number of items for the September meeting is too high and the Subcommittee believes it would be difficult for the Committee to complete the review of all these items during a 3-day meeting. Therefore, the Subcommittee proposes that the September meeting be held between Wednesday, September 5 and Saturday, September 8, 2001, instead of September 6-8 as currently scheduled.

3) Quadripartite Meeting Update

During the April meeting, the Committee was informed that recently Mr. Lothar Hahn, Chairman of the RSK, told us that preparations are being made by Germany to host the next Quadripartite meeting, possibly later this year. The French GPR have confirmed their participation and the RSK is currently working to confirm the participation of the Japanese NSC.

During the June 2001 meeting, the Committee proposed the following topics for the Quadripartite meeting:

- Risk-Informed Regulation
- Thermal-Hydraulic Analysis and Code Issues
- High Burnup Fuel

- Risk Analysis of Spent Fuel Storage

The Committee also suggested that other countries (e.g., Sweden and Switzerland) be invited to attend this meeting and that Dr. Larkins inform RSK about the Committee's suggestion. RSK has informed Dr. Larkins that they plan to discuss the ACRS suggestion with other Quadripartite member countries. Also, the RSK would prefer to have the Quadripartite meeting during the first full week in June 2002, but because of the anticipated conflict with the ACRS meeting, it suggests that the Quadripartite meeting be held on June 24-28, 2002 in Berlin, Germany.

RECOMMENDATION

The Subcommittee recommends the following:

- During the June meeting, the Committee agreed with Dr. Larkins' proposal that breakout sessions be planned to discuss topics other than the four main topics proposed by the Committee. The Committee should select a list of topics for the breakout sessions.

- The Committee should propose June 17-21, 2002, as possible dates for the Quadripartite meeting instead of June 24-28 proposed by RSK.

- The Committee should add "Human Performance" to the list of main topics for the meeting.

- Dr. Larkins should keep the Committee informed of the feedback from RSK on the proposed topics, participation by other countries, and any changes in the date for the Quadripartite meeting

4) Tour of the Shipyard in Groton, CT, and a Submarine

The ACRS plans to review the new nuclear propulsion plant submarine design (VIRGINIA Class, successor to the LOS ANGELES Class) in 2002. In connection with this review, the members visited the Naval Reactor (NR) Organization Headquarters Office in Crystal City, Virginia on April 4, 2000. On August 7, 2000, the members visited the NR training complex located at the Charleston, SC Naval Base. Recently, representatives of NR discussed with Dr. Apostolakis about potential options for the Committee's review of the VIRGINIA Class submarine as well as a tour of the shipyard construction site in Groton, CT, in November 2001 and tour of a submarine in early 2002. Mr. Sieber, Chairman of the Naval Reactors Subcommittee raised some issues with regard to the need for the Committee's sea voyage on a submarine (pp. 16-18). Traditionally, it has been the practice of the ACRS to tour a submarine in connection with its review of a new submarine design.

RECOMMENDATION

The Subcommittee recommends the following:

- The Committee members should visit the shipyard construction site in Groton, CT, in November 2001. The Subcommittee proposes November 19 (p.m.) -20, 2001 for this tour.

- The Committee should tour a submarine in early 2002.

5) Revised Subcommittee Structure

A revised ACRS Subcommittee structure (pp. 19-35) that was approved by Dr. Apostolakis, ACRS Chairman, was sent to all members on June 18, 2001, requesting comments by July 2, 2001. No comments were received.

RECOMMENDATION

The Subcommittee recommends that the Committee approve the revised Subcommittee structure and make it effective on July 16, 2001.

6) Member Requests for Support Services

Whenever members request the ACRS Office to set up an arrangement for support services (e.g., postage, storage space, rental of office space) at their off-site location and subsequently decide that they don't need the requested service(s), the members should inform us as soon as possible of their decision so that we may cancel the arrangement. Otherwise, money that could be better used to fund Office travel demands or other needs remains committed to provide that service and is lost beyond retrieval for the Office use after September 30 each year.

RECOMMENDATION

The Subcommittee recommends that the members provide timely notification to the ACRS Office when they decide that they do not need a requested service so that the contract arrangement may be terminated and the money returned to the budget to meet other needs.

7) Availability of Business Cards

The EDO has recently authorized the purchase of business cards for employees who perform representational duties requiring them to interact with or conduct NRC business and/or meetings with outside entities. The cards are printed in blue or black ink and there are two layout styles from which to choose. Card quantities of 250 or more must be ordered. We will print smaller orders in house on perforated business card stock.

RECOMMENDATION

Those members who would like to receive ACRS business cards should see Ethel Barnard to choose the layout, color, and quantity of cards.

8) Inadvertent Release of Documents to the Public

On June 19, 2001, the NRC discovered that approximately 800 documents stored in the ADAMS (Agency Wide Documents Access and Management System) Main Library marked as "non-public" were inadvertently made available to the public. Some of the documents were site access authorization letters from NRC to various licensees which contain privacy act information for certain NRC employees, including the members who have participated in site visits since 1999.

Members whose information was released have received a letter from the Executive Director of Operations, explaining how the problem came about and what remedial measures have been taken to preclude recurrence of this situation.

RECOMMENDATION

The Subcommittee recommends that if any members have questions or concerns regarding this issue, they should discuss it with the ACRS Executive Director.

9) Travel Request

Dr. Apostolakis has requested ACRS support to attend workshop on human reliability on October 8-10, 2001 in Munich, Germany hosted by the GRS and organized by the NEA (pp. 36-40). The workshop's objectives are stated as to exchange information on the technical issues associated with human reliability data needs. Issues include the availability of data to support modeling dynamic human performance, human aspects of common cause failures, latent errors especially those associated with maintenance, plant-specific operational experience and events, and simulator results.

RECOMMENDATION

The Subcommittee recommends that the Committee approve the travel request by Dr. Apostolakis.

10) Member Issue

Dr. Powers has proposed that a Subcommittee meeting be scheduled to review issues of seismic threat to nuclear power plants (pp. 41-43). A Task Group of about 3 ACRS members with mechanical engineering expertise should reexamine the IPEEE results for seismic issues, the recent DPV on Seismic PRA, and the NRC seismic research activities. Additionally, Dr. Powers suggests that the Committee notify the NRC staff that the ACRS will examine the draft Regulatory Guide 1.92 after reconciliation of public comments.

RECOMMENDATION

The Subcommittee recommends the following:

- A new Subcommittee named "Natural Phenomena" should be established. Dr. Powers will chair this Subcommittee. Members of this Subcommittee will be Dr. Apostolakis, Dr. Kress, Dr. Shack, and Mr. Rosen. This Subcommittee will review the IPEEE results for seismic issues, NRC seismic research activities, and other seismic-related issues.

- The differing professional view (DPV) on seismic PRA, which was sent to Dr. Powers, is expected to be resolved by the agency using the agency DPV resolution process. At this time, the Committee should not interfere with the agency efforts to resolve this DPV. The members, however, should read the DPV document (pp. 44-51) and understand the issues raised in the DPV.

11) Items of Interest

- We issued the ACRS/ACNW Operating Plan to the Commission on June 28, 2001, satisfying a Commission milestone. Copies of the Operating Plan had been sent to the ACRS and ACNW members. We committed to provide an updated Operating Plan and the ACRS/ACNW letter matrix to the Commission in December 2001.

- The Commission has appointed Mr. Rosen to the ACRS effective June 13, 2001.

- In a Staff Requirements Memorandum (SRM) dated June 12, 2001, (pp. 42-43), authorizing the NRR Director to renew the operating license for ANO Unit 1, the Commission commended the ACRS for the outstanding efforts associated with the thorough and timely review of the ANO Unit 1 license renewal application.

- A list of upcoming NEI events is attached (p. 52) for the members information. Members needing additional information should contact Dr. Larkins or Dr. Savio.

ANTICIPAT WORKLOAD
July 11-13, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Singh	Draft IPEEE Insights Report	Report	RPRA 6/22 P&P 7/10 (p.m.)	PO/FP 6/27-28 M&M/THP/RPRA 7/10 PO 7/9
Bonaca	Ford	Duraiswamy/Dudley	Need for Revising 10 CFR Part 54-The License Renewal Rule	Report		RPRA 6/22 M&M/THP/RPRA 7/10 P&P 7/10 (p.m.) PO 7/9
Ford	Sieber	Weston/Boehnert	Staff and Industry proposals for dealing with Control Rod Drive Mechanism Cracking	Report	M&M/PO 7/10	RPRA 6/22 PLR 6/22 PO/FP 6/27-28
Kress	--	El-Zeftawy	Spent Fuel Accident Risk at Decommissioning Plants	Report	--	THP 6/12 RPRA 6/22 M&M/THP/RPRA 7/10 PO 7/9 P&P 7/10 (p.m.)
Leitch	Bonaca	Singh	Proposed Resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance **[STATUS REPORT]**	--		PO/FP 6/27-28 PO 7/9

ANTICIPATED WORKLOAD
July 11-13, 2001 (CONTINUED)

Sieber	Leitch Apostolakis	Weston	South Texas Project Exemption Request Reactor Oversight Process **[Committee Discussion]**	Report Draft Report	PO/FP 6/27-28 PO 7/9	--
Shack	--	Markley	Proposed Risk-Informed Revisions to 10 CFR 50.46 and Revisions to the Framework for risk-informing the Technical Requirements of 10 CFR Part 50	Report	M&M/THP/RPRA 7/9	THP 6/12 RPRA 6/22
Wallis	Bonaca	Boehnert/Cronenberg	Potential Margin Reductions Associated with Power Uprates	Report (Tentative)	THP 6/12	M&M/THP/RPRA 7/10 (p.m.) PO 7/9

ANTICIPATED WORKLOAD
SEPTEMBER 6-8, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Larkins	Meeting with Commissioner Merrifield	--	P&P 9/4 (3:00p.m.) (Video conferencing)	
Bonaca	--	Duraiswamy	Proposed Update to 10 CFR Part 52	Report	--	P&P 9/4 (p.m.)
Rosen		Singh	Proposed Resolution of GSI-191, Assessment of Debris Accumulation on PWR Sump Performance	Report		THP 7/17-18 THP 8/21-23
Kress		El-Zeftawy	EPRI Report on Waterhammer Issues	Report	THP 8/21-23	THP 7/17-18
		El-Zeftawy	Safeguards Issues Associated with Spent Fuel fire risk at Decommissioning Plants	Report		P&P 9/4 (p.m.)
Powers		Boehnert	Duane Arnold Core Power Uprate	Report	--	THP 7/17-18 THP 8/21-23
		Dudley	Proposed Revision to Reg. Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"	Report		
Sieber	Apostolakis	Weston	Reactor Oversight Process	Report	--	--
Wallis		Boehnert	TRACG Best-Estimate Thermal-Hydraulic Code	Report	THP 7/17-18	

ANTICIPATED WORKLOAD
OCTOBER 4-6, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Markley	Risk-Informed Regulation Implementation Plan	Report (Tentative)	P&P 10/3	
Bonaca	Leitch	Dudley	Interim Review of the License Renewal Application for Turkey Point Units 3 and 4	Report	PLR 9/24-25	P&P 10/3
Kress	--	Dudley	Proposed Regulatory Guides on Control Room Habitability	Report	--	THP 9/26-27 PLR 9/24-25 P&P 10/3
Powers	Ford	Duraiswamy	Action Plan on Steam Generator DPO Issues	Report	--	THP 9/26-27
Wallis	Sieber	Boehnert	Dresden and Quad Cities Core Power Uprate	Report	THP 9/26-27	--
			Proposed Reg. Guide and SRP Associated with NRC Code Reviews	Report		
			RETRAN-3D Transient Analysis Code	Report		

10

II. ITEMS REQUIRING COMMITTEE ACTION

1. Steam Generator Tube Integrity Issues (Open) (FPF/NFD) ESTIMATED TIME: 2 hours

   Purpose: Determine a Course of Action

   Review requested by the NRC staff. [R. Ennis, NRR] The proposed generic letter and Regulatory Guide associated with steam generator tube integrity were reviewed by the Committee to Review Generic Requirements on July 21, 1998. In a memorandum dated September 11, 1998 to the EDO, the staff proposed to delay issuance of the proposed generic letter for three months while it worked with industry to reach agreement on the content of industry guidelines. The staff issued the draft regulatory guide, "Steam Generator Tube Integrity," and the differing professional opinion (DPO) response for public comment on January 20, 1999. The Materials and Metallurgy Subcommittee heard a status briefing on this issue at its March 24-25, 1999 meeting. The staff plans to resolve GSI-163, "Multiple Steam Generator Tube Leakage," on the basis of the information contained in the DPO resolution package. The Committee issued a letter on April 22, 1999, regarding the status of resolution of steam generator tube integrity issues.

   On February 16, 2000, the staff issued a Steam Generator Action Plan that contained steam generator Action Plan milestones, non-steam generator related Action Plan milestones, and Indian Pont Unit 2 Task Group Recommendations. The staff updated the Action Plan in March and April 2001. The Materials and Metallurgy Subcommittee plans to hold a meeting in September 2001, to discuss the status of the steam generator tube integrity issues and action plan milestones.

   The Planning and Procedures Subcommittee recommends that following the Subcommittee meeting, Dr. Ford propose a course of action for reviewing this matter.

2. Draft Regulatory Guide DG-1077 on Environmental Qualification of I&C Equipment (REU/SR/AS) ESTIMATED TIME: 1 hour

   Purpose: Determine a Course of Action

   Review requested by NRC staff [S. Arndt]. Oak Ridge National Laboratory (ORNL) completed a draft NUREG on the proposed qualification methodology of advanced instrumentation and control (I&C) based on endorsement of two standards (IEEE 323 and IEC-60780). DG-1077, "Guidelines for Environmental Qualification of Microprocessors-Based Equipment Important to Safety in Nuclear Power Plants," is based on the information contained in the NUREG. The staff provided the ACRS with copies of DG-1077 and the NUREG on June 8, 2001, and recommends that the ACRS review this matter after the resolution of public comments.

//

The Planning and Procedures Subcommittee recommends that the Committee consider reviewing this Guide after reconciliation of public comments and that Dr. Uhrig and Mr. Rosen provide their views.

3. Review of Draft Regulatory Guide DG-1108,, "Combining Modal Responses and Spatial Components in Seismic Response" (DAP/NFD) ESTIMATED TIME: 1 hour

Purpose: Determine a Course of Action

Review requested by NRC staff [M. Mayfield]. Regulatory Guide 1.92, "Combining Modal Responses and Spacial Components in Seismic Response," Revision 1, was issued in February 1976. Advances in this area over the last 25 years, including NUREG/CR-6645, "Re-evaluation of Regulatory Guidance on Modal Response Combination Methods for Seismic Response Spectrum Analysis," have not been incorporated into the Regulatory Guide. The staff provided the ACRS with copies of DG-1108, which is a proposed revision to RG-1.92, and NUREG/CR-6645 on June 1, 2001, and suggested that the ACRS review the draft regulatory guide after public comments have been reconciled.

Dr. Powers recommends that the Committee review the proposed final version of this Guide after reconciliation of public comments. Also, a Subcommittee meeting should be scheduled to discuss the issue of Seismic threats to nuclear power plants. In addition, a Task Group of three ACRS members with mechanical engineering background should be established to reexamine the IPEEE results for seismic issues and the NRC seismic research activities to identify a list of questions for the staff to address in its presentation to the ACRS.

4. Review of DOE/DOD Naval Reactors Virginia Class Nuclear Propulsion Plant Submarine Design (JDS/PAB) ESTIMATED TIME: 2 hours

Purpose: Determine a Course of Action

DOD/DOE Naval Reactors Review Request. [A. Adams, NRR]. The Naval Reactors (NR) Organization will be submitting documentation pertaining to its new nuclear propulsion plant (NPP) submarine design (VIRGINIA Class, successor to the LOS ANGLES Class) to the NRC and ACRS for review in early-July 2001. The Committee last reviewed an NR NPP plant design (SEAWOLF) in 1994. Only three of the current ACRS members were on the Committee at the time of that review.

Dr. Powers had suggested that the Committee interact with NR, early on, to become familiar with its organization, history, and approach. The Committee Members visited the NR Headquarters Office at Crystal City, Virginia and discussed the Naval Reactors program on the morning of April 4, 2000. Committee Members also visited the NR training complex located at the Charleston, South Carolina Naval Base on August 7, 2000. This complex is

comprised of the Moored Training Ships and the Nuclear Power Training School.

Recently, a NR representative held a discussion with Dr. Apostolakis regarding scheduling and potential options for the Committee's review of the VIRGINIA NPP. Among the specifics discussed included a tour of the shipyard construction site in Groton, CT this November. It was also proposed that the Committee tour a nuclear powered submarine, probably in early-2002. Dr. Apostolakis has suggested that the Committee discuss this matter during the July Planning & Procedures Subcommittee and the July Committee Meeting.

The Committee will be kept informed of the schedule milestones for this review. NR is scheduled to submit the SSAR and PRA documents associated with the VIRGINIA NPP to the staff in July 2001. Final review of this matter by the ACRS is now expected in the August/September 2002 timeframe.

The Planning and Procedures Subcommittee recommends that the Committee tour the shipyard construction site in Groton, CT, on November 19 (p.m.)-20, 2001 and also tour a submarine in early 2002.

5. SECY-01-0094,"Staff Review of Request for Exemptions and an Associated Amendment Related to Physical Security Plans for an Independent Spent Fuel Storage Installation (ISFSI) Under General License" (Open) (TSK/MME)
ESTIMATED TIME: 1 hour

Purpose: Determine a Course of Action

Briefing requested by the ACRS/NRC [Vonna Ordaz, NRR]. The staff prepared the subject SECY to inform the Commission of the staff's approach for reviewing the Maine Yankee Atomic Power Company request for exemptions and an associated amendment related to physical security for ISFSI. The staff is also informing the Commission of a proposed rulemaking that would allow 10 CFR Part 50 licensees to design ISFSI security plans in accordance with 10 CFR 73.51. The subject SECY outlines the staff's potential for long-term activities related to the current ISFSI physical security requirements.

The Planning and Procedures Subcommittee recommends that Dr. Kress propose a course of action.

6. SECY-01-0101,"Proposed Rule Changes to 10 CFR 73.55: Requirements for Physical Protection of Licensed Activities at Nuclear Power Reactors against Radiological Sabotage; 10 CFR Part 72: Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste; And 10 CFR 50.54(p): Conditions of Licenses" (Open) (TSK/MME)
ESTIMATED TIME: 1 hour

Purpose: Determine a Course of Action

Briefing requested by the NRC Staff [R. Rosano, NRR]. The NRC is proposing to amend its physical protection requirements to provide a more performance-based approach for nuclear power reactor security programs and to employ risk insights when determining plant systems to be protected against the design basis threat (DBT). The proposed rule would require licensees to revise current onsite security programs and organizations while maintaining the objective of providing high assurance that licensed activities at nuclear power plants do not constitute an unreasonable risk to public health and safety as a result of radiological sabotage by the DBT.

The staff is seeking Commission approval to publish the subject SECY in the *Federal Register* for a 120-day public comment period.

The Planning and Procedures Subcommittee recommends that Dr. Kress propose a course of action.

7. Control Room Habitability (Open) (TSK/NFD) ESTIMATED TIME: 1 ½ hours

Purpose: Determine a Course of Action

Review requested by the ACRS. [J. Hayes, NRR] The NRC staff initiated an effort to resolve issues associated with control room habitability, primarily due to problems with uncontrolled inleakage. The Nuclear Energy Institute (NEI) developed a draft guidance document, NEI 99-03, "Control Room Habitability Assessment Guidance," in August 1999. The staff identified significant concerns with this draft document. The Severe Accident Management Subcommittee reviewed this matter during its September 16-17, 1999 meeting, and the Subcommittee Chairman provided a report to the ACRS during the October 1999 meeting.

NEI and the NRC staff met on January 13, 2000, and setup NRC/Industry Subgroups to resolve the open issues. The ACRS Severe Accident Management Subcommittee met on November 15, 2000 to review the status of this work. The full Committee reviewed and commented on a revised NEI 99-03 during the December 2000 meeting. NEI is sponsoring a Workshop on control room habitability issues on August 23-24, 2001 in Clear Beach, Florida.

The staff is developing four regulatory guides concerning control room habitability, dose assessment, meteorological effects, and in-leakage testing. These guides are intended to also provide guidance to advanced reactor applicants on the design of main control rooms. The staff plans to issue these regulatory guides in conjunction with a generic letter. The staff plans to provide copies of the regulatory guides and the generic letter to the Committee by August 31, 2001.

The Planning and Procedures Subcommittee recommends that the Committee consider reviewing the proposed final version of these Guides after reconciliation of public comments. After receiving copies of these

Guides, a Larkinsgram should be issued to inform the EDO of the Committee decision.

8. Proposed Update to 10 CFR Part 52 (Open) (MVB/SD) ESTIMATED TIME: 1½ hours

Purpose: Determine a Course of Action

Review requested by the NRC staff [J. Wilson, NRR] Based on the insights gained from the review of the ABWR, CE-System 80+, and AP600 designs, and the comments received on 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants," the staff is in the process of updating 10 CFR Part 52. This update will take into account experience from the previous design certification rulemakings and will update and correct the licensing processes in 10 CFR Part 52 to prepare for future applications. The staff expects to provide the document to the ACRS in August 2001 and brief the full Committee at the September 2001 ACRS meeting.

Dr. Bonaca will provide his views on the need for the Committee to review this matter.

9. Status of NRC Programs for Nuclear Power Plant Safeguards and Security (Open)(GML/NFD) ESTIMATED TIME: 1 hour

Purpose: Determine a Course of Action

Review requested by the ACRS [V. Ordaz, NRR]. In the proposed physical security rulemaking plan (SECY-99-241), the staff recommended that the NRC conduct a comprehensive review of its power reactor security regulations, including the requirement for licensees to conduct drills and exercises to evaluate the effectiveness of their response capabilities to safeguards contingency events. In SECY-99-241, the staff discussed a proposal by the Nuclear Energy Institute (NEI) to develop a pilot program for industry-conducted drills and exercises. In an SRM dated November 22, 1999, the Commission approved the staff's recommendations. On April 13, 2001, the staff provided its draft Commission paper on the Safeguards Performance Assessment (SPA) Pilot Program for consideration by the Committee.

During the April 2000 ACRS meeting, the Committee decided not to review the SPA Pilot Program. However, Dr. Kress recommended and the Committee agreed that a Subcommittee meeting be scheduled in mid-2001 to review the status of NRC programs for nuclear power plant safeguards and security. A Subcommittee meeting is expected to be held in September 2001, to review this matter.

The Planning and Procedures Subcommittee recommends that subsequent to the Subcommittee meeting, Mr. Leitch recommend whether a briefing to the full Committee on this matter is needed.